

***A Comprehensive Inventory of
Radiological and
Nonradiological Contaminants
in Waste Buried or Projected to
be Buried in the Subsurface
Disposal Area of the INEEL
RWMC during the Years of
1984 to 2003 Supplemental
(Volume 1 of 2)***

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August 2001

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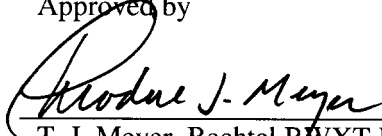
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during the Years 1984 to 2003 Supplement**

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Approved by



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Date

ABSTRACT

This report is a supplement to the 1995 document *A Comprehensive Inventory of Radiological and Nonradiological Contaminants in Waste Buried or Projected to Be Buried in the Subsurface Disposal Area of the INEL RWMC during the Years 1984–2003*.

This supplemental report documents an update to the comprehensive inventory of radiological and nonradiological contaminants that were shipped from 1994 to 1999 from various Idaho National Engineering and Environmental Laboratory facilities to the Radioactive Waste Management Complex and then buried in the Subsurface Disposal Area. The update involved replacing projected inventories with actual inventory data or improved estimates. These updated inventories have been compiled primarily to perform the baseline risk assessment under the Comprehensive Environmental Response, Compensation, and Liability Act. The baseline risk assessment will support the Operable Unit 7-13/14 comprehensive remedial investigation/feasibility study.

The methodology used to identify, collect, compile, review, and enter waste inventory information into the Contaminant Inventory Database for Risk Assessment is described in detail. The source documents used to compile the information also are described. In addition, descriptions are provided of (1) the facilities shipping the waste (i.e., the waste generators), (2) the processes by which the waste was generated, (3) the availability of information, (4) the sources of data, and (5) the approach used to collect data.

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ACRONYMS

ANL-E	Argonne National Laboratory-East
ANL-W	Argonne National Laboratory-West
ARA	Auxiliary Reactor Area
ATR	Advanced Test Reactor
CERCLA	Comprehensive Environmental Response, Compensation and Liability Act
CFA	Central Facilities Area
CIDRA	Contaminant Inventory Database for Risk Assessment
D&D&D	deactivation, decontamination, and decommissioning
DOE	U.S. Department of Energy
EBR-I	Experimental Breeder Reactor I
EBR-II	Experimental Breeder Reactor II
ECF	Expended Core Facility
EPRI	Electric Power Research Institute
ER	environmental restoration
FCF	Fuel Cycle Facility
FMF	Fuel Manufacturing Facility
HDT	historical data task
HEPA	high-efficiency particulate air
HFEF	Hot Fuel Examination Facility
INEEL	Idaho National Engineering and Environmental Laboratory
INTEC	Idaho Nuclear Technology and Engineering Center
IWTS	Integrated Waste Tracking System
L&O	Laboratory and Office Building
LLW	low-level waste
LOFT	loss-of-fluid test

NRF	Naval Reactors Facility
NPR	New Production Reactor
ORIGEN2	Oak Ridge Isotope GENeration and Depletion Code Version 2
OU	operable unit
PBF	Power Burst Facility
PWR	pressurized water reactor
RCRA	Resource Conservation and Recovery Act
RLWTF	Radioactive Liquid Waste Treatment Facility
RPDT	recent and projected data task
RSD	relative standard deviation
RWMC	Radioactive Waste Management Complex
RWMIS	Radioactive Waste Management Information System
SCMS	Sodium Components Maintenance Shop
SDA	Subsurface Disposal Area
SF	scaling factor
SMC	Specific Manufacturing Capability
SPF	Sodium Process Facility
SPERT	Special Power Excursion Reactor Test
TAN	Test Area North
TMI	Three Mile Island
TRA	Test Reactor Area
TREAT	Transient Reactor Test Facility
TRISO	tri-isotropic
TRU	transuranic
TSF	Technical Support Facility
WAG	waste area group

WERF	Waste Experimental Reduction Facility
WMF	Waste Management Facility
WROC	Waste Reduction Operations Complex
ZPPR	Zero Power Physics Reactor

A Comprehensive Inventory of Radiological and Nonradiological Contaminants in Waste Buried or Projected to Be Buried in the Subsurface Disposal Area of the INEEL RWMC during the Years 1984 to 2003 Supplement

1. INTRODUCTION AND BACKGROUND

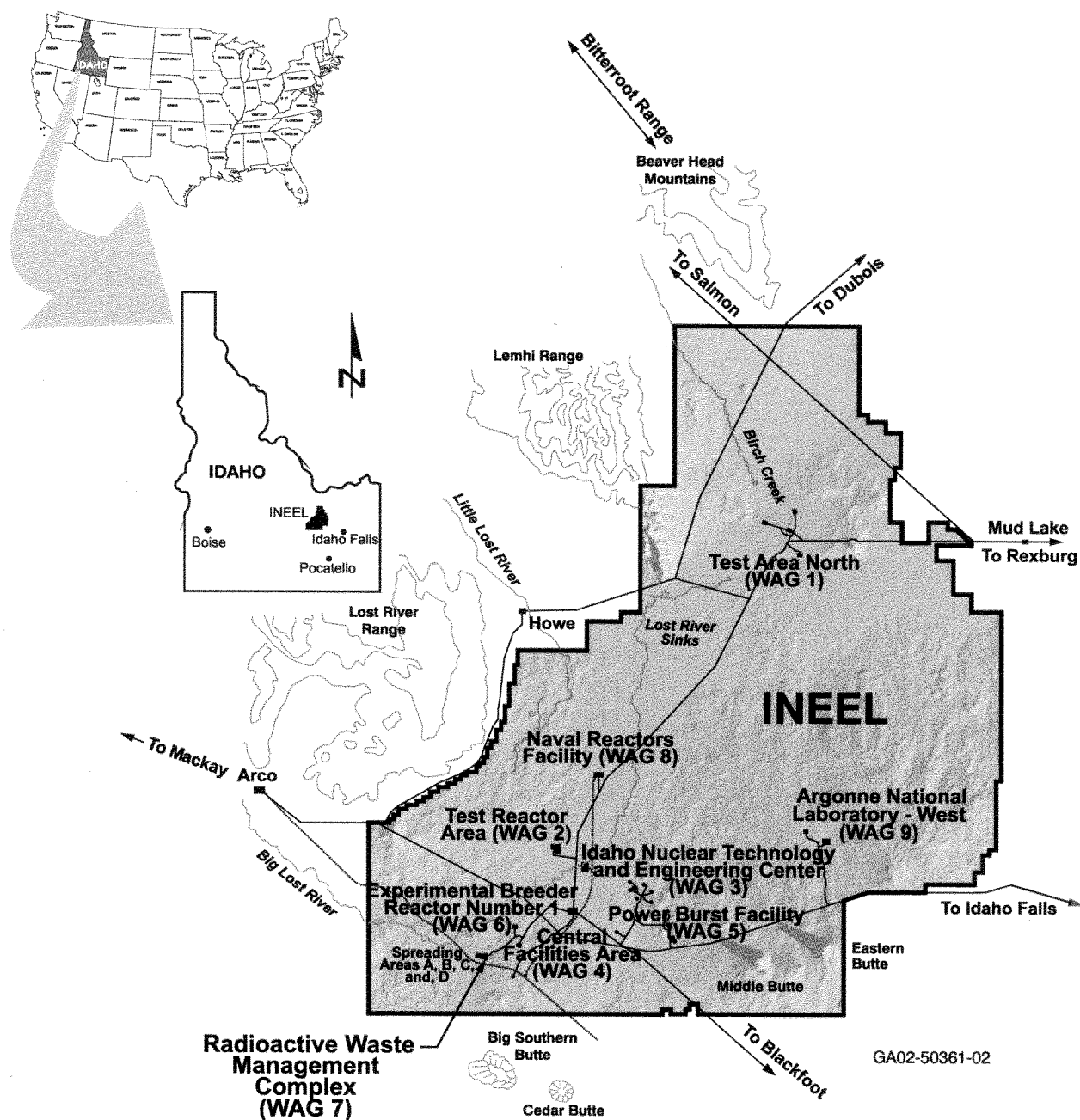
1.1 Objective and Overview

This report is a supplement to the 1995 document *A Comprehensive Inventory of Radiological and Nonradiological Contaminants in Waste Buried or Projected to Be Buried in the Subsurface Disposal Area of the INEL RWMC during the Years 1984–2003*, which is commonly known as the Recent and Projected Data Task (RPDT) (LMITCO 1995a). This supplement documents an update to the comprehensive inventory of radiological and nonradiological contaminants that were shipped from 1994 to 1999 from various Idaho National Engineering and Environmental Laboratory (INEEL) facilities to the Radioactive Waste Management Complex (RWMC) and then buried at the Subsurface Disposal Area (SDA). The updated inventory involved replacing projected inventory with actual inventory data or improved estimates. These waste inventories have been compiled primarily to perform a baseline risk assessment for the Operable Unit (OU) 7-13/14 comprehensive remedial investigation/feasibility study required by the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) (42 USC § 9601 et seq.). The location of the RWMC at the INEEL is shown in the contour map in Figure 1-1.

The CERCLA applies to hazardous waste, in accordance with the Resource Conservation and Recovery Act (RCRA) (42 USC § 6901 et seq.), and other hazardous substances. Operable Unit 7-13/14 is the designation for the comprehensive OU for Waste Area Group 7, which comprises the RWMC, recognized under the Federal Facility Agreement and Consent Order (FFA/CO) (DOE-ID 1991) and CERCLA.

This supplement is a compilation of a comprehensive inventory of radiological and nonradiological contaminants shipped from the facilities (i.e., waste generators) or areas and buried in the SDA from 1994 to 1999.

- Argonne National Laboratory-West (ANL-W)
- Test Reactor Area (TRA)
- Test Area North (TAN)
- Specific Manufacturing Capability (SMC) Facilities. Note that while the SMC facilities are part of TAN, SMC has been listed and evaluated separately because it has a unique waste stream consisting of depleted uranium.
- Idaho Nuclear Technology and Engineering Center (INTEC), formerly known as the Idaho Chemical Processing Plant
- Naval Reactors Facility (NRF)



WAG 10 includes all sites, disposal areas, and portions of the Snake River Plain Aquifer that either are outside the boundaries of WAGs 1 through 9 or are not included within the other WAGs.

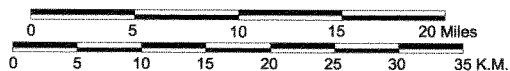


Figure 1-1. Contour map of the Idaho National Engineering and Environmental Laboratory showing the location of the Radioactive Waste Management Complex and other major facilities.

- Other facilities and areas. The results for some facilities have been grouped under the heading of “other waste generators” because of the small amounts of contamination in the waste streams. Also, note, as indicated below, that waste area groups (WAGs) are listed separately from the corresponding facilities because WAGs are waste generators through the CERCLA process and generate different types and amounts of waste than the facilities.
 - Central Facilities Area (CFA)
 - Power Burst Facility (PBF)
 - Waste Experimental Reduction Facility (WERF)
 - Auxiliary Reactor Area (ARA)
 - Radioactive Waste Management Complex
 - Waste Area Group (WAG) 1
 - Waste Area Group 3
 - Waste Area Group 7
 - Deactivation, decontamination, and decommissioning (D&D&D).

Disposal locations within the SDA are not included in this supplement.

1.2 Brief History and Description of the Subsurface Disposal Area

See the description of the SDA in the RPDT.

1.3 Document Organization

A brief description of the remaining sections in this update follows:

- Section 2—The methods used to update the radionuclide and nonradionuclide waste inventory information for each facility shipping waste disposal to the SDA from 1994 to 1999 are described
- Section 3—The results of the inventory update are presented.
- Section 4—References cited throughout the document are listed.

2. METHODOLOGY TO COLLECT AND COMPILE DATA

The methods used to update the waste inventory information from facilities shipping radionuclide and nonradionuclide waste disposals to the SDA from 1994 to 1999 are described in Section 2. The process included identifying, collecting, compiling, reviewing, and entering waste inventory information into a central database. Methods used in this report were the same as those used in the RPDT for the period 1984 through 1993.

2.1 Overview

The Radioactive Waste Management Information System (RWMIS) database served as the starting point to update waste disposals at the SDA between 1994 and 1999. In June 1997, waste-tracking databases across the INEEL were combined into a central database called the Integrated Waste Tracking System (IWTS) (see Section 2.2). Because the quality of information in the RWMIS and IWTS databases has improved gradually over the years, updating information, in most cases, was less difficult than in previous years. The first step in compiling the data for this supplement involved extracting data from the IWTS database for each facility under consideration. This step proved useful because each of the various facilities has its own unique waste stream for which different analysis methodologies are useful, depending on the available information. The following facilities (i.e., waste generators) and areas were evaluated for this inventory update:

- Argonne National Laboratory-West
- Test Reactor Area
- Test Area North
- Specific Manufacturing Capability
- Idaho Nuclear Technology and Engineering Center
- Naval Reactors Facility
- Other facilities and areas
 - Central Facilities Area
 - Power Burst Facility
 - Waste Experimental Reduction Facility
 - Auxiliary Reactor Area
 - Radioactive Waste Management Complex
 - Waste Area Group 1
 - Waste Area Group 3
 - Waste Area Group 7
 - Deactivation, Decontamination, and Decommissioning.

The waste from each facility or waste generator was characterized by dividing the waste from a specific facility into waste streams. A waste stream is defined as a collection of waste containers with similar contents. The waste streams from each of the facilities are listed in following sections. Updated information from each of the waste streams (both radiological and nonradiological) was entered into the Contaminant Inventory Database for Risk Assessment (CIDRA) database on a yearly basis (LMITCO 1995a, 1995b). This information included the facility (i.e., waste generator), building, and assigned number of waste streams from that building. It also included the waste volume, physical and chemical form of the waste, nonradiological and radiological contaminant quantities, verbal descriptions, and comments. The total required information is listed in Appendix A, which is a standardized five-page data form previously used to compile and record waste data.

Each waste stream was updated by using scaling factors to calculate radioisotope curie amounts for certain nuclides not already listed in the IWTS database. The particular set of scaling factors used for each waste stream depended on the information available for that waste stream. In some cases (i.e., for TRA resins and SMC depleted uranium), recent analytical data that included estimated measurement errors were available from the Analytical Laboratory Department at TRA and INTEC. Using standard statistical analysis, scaling factors and upper-bound scaling factors (SF^{up} s) could be calculated using these data. In other cases (such as for ANL-W), computer calculations using the Oak Ridge Isotope GENERation and Depletion Code Version 2 (ORIGEN2) code (Croff 1980) were used to estimate scaling factors. Scaling factors were derived from data in the RPDT or data from the Electric Power Research Institute (EPRI) for particular radioisotopes where data were not otherwise available (Best and Miller 1987). The EPRI data were gathered from commercial nuclear power operators.

The activity of radionuclides expected to be present, but not reported in the RWMIS database (or in other records) concerning the waste shipments made to the RWMC, were computed based on other parameters. Unreported radionuclide inventories were computed based on scaling factors and the reported curie inventories of certain reference isotopes. The three reference isotopes selected for this investigation were Co-60, Cs-137, and Pu-239. Cobalt-60 was used as the reference isotope for all activation products, Cs-137 was used as the reference isotope for fission products, and Pu-239 was used as the reference isotope for transuranic (TRU) waste or actinides. The scaling-factor data were determined from either theoretical calculations or previously reported information and were dependent on the specific facility that generated the particular waste. For example, the scaling factor for Mn-54 is known for the ANL-W facility (e.g., activity ratio of Mn-54/Co-60), then the Mn-54 inventory in an ANL-W waste shipment can be computed by multiplying the Mn-54 scaling factor by the activity of Co-60 that was reported for this waste shipment.

Curie inventories were so small in the case of facilities listed as other waste generators that attempts to update them with scaling factors were not considered. Upper bounds were calculated using the generic relative standard deviations (RSDs) found in the RPDT (LMITCO 1995a, Section 5.4.3). These generic numbers are derived from EPRI data.

2.2 Source Documents and Information

The IWTS, formerly known as the RWMIS database, was the primary source used to compile this inventory. The environmental database integration project was initiated in June 1997 to combine waste tracking systems across the INEEL into a single reporting system (DOE-ID 1998). Historical information for RWMIS was loaded into a data repository and all previous reporting capabilities became available in IWTS, along with additional detailed information for tracking containerized waste.

Several other sources of information also were used to compile and verify the inventory. These sources included process knowledge, operating logs, previous inventory-related documents, shipping

records, information databases, waste-generator forecasts, engineering and nuclear physics calculations, and interviews with personnel having knowledge of the facility operations that produced the waste streams. The inventory was compared with those in other reports and databases to confirm completeness, and the reasons for noted differences were explored. Those differences were noted and explained on the inventory data sheets (see Appendix B).

2.3 Data Collection Methods

The methods used to collect waste information for each of the seven waste generators listed previously in Section 2.1 are discussed in this section. The waste streams discussed below varied not only from facility to facility but from one stream to another. In addition, the data-collection methods also differed from stream to stream. Therefore, in general terms, (1) each of the seven waste generators, (2) processes by which waste was generated, (3) availability of information, and (4) the data-collection approach used are described in Sections 2.3.1 through 2.3.7.

2.3.1 Argonne National Laboratory–West

2.3.1.1 Waste Generator. Argonne National Laboratory–West is located in the southeastern part of the INEEL, approximately 56 km (35 mi) west of Idaho Falls, Idaho. It is the prime testing center in the United States for nuclear energy technology demonstration and proof-of-concept. The mission at ANL-W emphasizes technologies associated with nuclear fuel, including advanced fuel treatment methods, fuel efficiency enhancements, and fuel performance testing. This mission also includes nuclear material characterization technologies, environmental technologies, and technologies and processes requiring remote handling of nuclear fuel. The designation recognized under the FFA/CO and CERCLA for ANL-W is WAG 9 (see Figure 1-1).

The following seven major complexes compose ANL-W:

- Experimental Breeder Reactor II (EBR-II)
- Transient Reactor Test Facility (TREAT)
- Zero Power Physics Reactor (ZPPR)
- Hot Fuel Examination Facility (HFEF)
- Fuel Cycle Facility (FCF)
- Fuel Manufacturing Facility (FMF)
- Laboratory and Office Building (L&O).

Support facilities at ANL-W include the following:

- Radioactive Liquid Waste Treatment Facility (RLWTF)
- Sodium Components Maintenance Shop (SCMS)
- Sodium Process Facility (SPF) (ANL-W 2001).

The EBR-II facility consists of (1) a sodium-cooled reactor with a thermal power rating of 62.5 MW, (2) an intermediate closed loop of secondary sodium, and (3) a steam plant that produces

19 MW of electrical power through a conventional turbine generator. The original EBR-II was designed to demonstrate a complete operational breeder reactor power plant with on-Site reprocessing of metallic fuel. The demonstration was successfully carried out from 1964 to 1969.

The emphasis at EBR-II then shifted to irradiation testing of fuels and materials for future, larger, liquid-metal reactors. The EBR-II facility also provided electrical power for ANL-W and the INEEL. The facility was shut down in June 1997. Since then, EBR-II has been prepared for D&D&D.

The TREAT reactor is a uranium-oxide-fueled, graphite-moderated, air-cooled reactor. It was designed to produce short, controlled bursts of nuclear energy to simulate accident conditions leading to nuclear fuel damage. The reactor became operational in 1959. Tests at TREAT provided data on fuel cladding damage, fuel motion, coolant channel blockages, molten fuel and coolant interactions, and potential explosive forces during an accident. Currently, the reactor is on standby status (ANL-W 2001).

The ZPPR is the national facility for testing the physics properties of advanced, fast-spectrum reactors. The ZPPR is designed to study the properties of experimental reactor cores. Experimental cores are built by hand loading plates of reactor materials into drawers that are then put into the designed pattern. The designs are tested at essentially zero power levels to determine characteristics of the cores. The ZPPR is now on standby status (ANL-W 2001).

The FCF (formerly called the Hot Fuel Examination Facility/South) became operational in 1964 and was used to demonstrate pyrometallurgical fuel reprocessing for EBR-II fuel during the first few years of operation. A remotely operated production line was used to process and refabricate spent EBR-II fuel and return it to the reactor. That mission was discontinued after successful demonstration of the process. Next, the facility was used to examine irradiated fuels and material experiments from EBR-II and TREAT, and to provide other reactor-support services such as spent fuel transfer to INTEC. The FCF consists of two hot cells, one with an air atmosphere and the other with an inert argon-gas atmosphere. A total of 23 hot cell workstations surround the outside perimeter of the FCF hot cells, and four active workstations make up the work space of the argon cell (ANL-W 2001).

The Hot Fuel Examination Facility, formerly the Hot Fuel Examination Facility/North, went into operation in 1975 and is used to examine irradiation experiments. Examinations conducted in the HFEF provide data essential for determining the performance and conditions of fuels and materials irradiated in the EBR-II reactor, the TREAT reactor, and other U.S. Department of Energy (DOE) reactor facilities. The HFEF consists of two shielded hot cells: (1) the decontamination cell, which contains an air atmosphere, and (2) the main cell, which contains an argon-gas atmosphere. Each of the 21 work stations in the HFEF is equipped with shielded windows and master/slave manipulators. The main cell is used for work involving exposure of materials such as sodium, plutonium, and other materials that would react chemically with air (ANL-W 2001).

The Neutron Radiography Facility, a 250-kW training, research, and isotope reactor is located in the basement of the HFEF and provides a neutron source for radiography. The Neutron Radiography Facility is equipped with two beam tubes and two separate radiography stations for neutron radiography of irradiated components. Facilities to decontaminate and repair hot-cell equipment and manipulators are also located within the HFEF (ANL-W 2001).

The capability to examine and characterize contact-handled TRU waste (destined for the Waste Isolation Pilot Plant in New Mexico) was added to HFEF in 1990 (ANL-W 2001).

The FMF contains the entire operation for manufacturing metallic fuel elements within a single building. The building contains a casting furnace and large gloveboxes to encapsulate and bond the cast-fuel slugs in a stainless steel jacket (ANL-W 2001).

The analytical laboratory in the L&O Building provides chemistry support for ANL-W in the areas of environmental compliance, fuel chemistry, sodium and water chemistry, and waste classification analysis. The laboratory consists of hot cells and chemistry laboratories.

Also associated with EBR-II, the SCMS support facility is used to remove sodium from reactor components for repair or replacement. Another support facility, the SPF, converts primary and secondary coolant from EBR-II from its elemental, chemically unstable form to a chemically stable compound suitable for landfill disposal (ANL-W 2001).

The RLWTF began operating in June 1983. The RLWTF receives low-level radioactive liquid waste from ANL-W facilities and stores the waste in storage tanks before evaporation in the shielded hot-air drum evaporators. Before 1983, the low-level liquid evaporation process took place in the basement of the L&O Building. Currently, liquid waste is piped from the L&O Building, the FCF, and the HFEF directly to the RLWTF facility (ANL-W 2001). The EBR-II cooling tower also is associated with EBR-II.

2.3.1.2 Generation of Waste. Low-level waste generated at ANL-W for disposal at the SDA between 1994 and 1999 inclusive consisted of contact-handled and remote-handled waste. Contact-handled LLW was generated in the FMF, L&O Building, FCF, HFEF, TREAT, ZPPR, EBR-II, SCMS, and RLWTF facilities. The waste generated by routine facility operations, maintenance, monitoring, and modifications included scrap metal, nonprocessable combustible waste, shade units, brick and concrete, and absorbed liquids. In addition to waste generated from routine operations, a small amount of one-time waste was generated from (1) crushed radioactively contaminated mercury lights and (2) treated NaK from EBR-I. Crushed radioactively contaminated mercury light waste was generated as a treatability study that included crushing radioactively contaminated mercury lights and stabilizing the residue in chemically bonded phosphate ceramics. The NaK was removed from Experimental Breeder Reactor I (EBR-I) in 1955 and stored until the mid-1990s. At that time, the NaK was treated and shipped to the SDA for disposal (WGS 1996).

The majority of the radioactivity (curies) in ANL-W waste disposed of at the SDA from 1994 to 1999 can be attributed to a shipment of one container from HFEF. However, the volume from the shipment represents less than 0.1% of the total volume of waste disposed of at ANL-W. The contact-handled waste from the remediation of the EBR-II leach pit, L&O, FCF, and HFEF contributed approximately 72% of the total volume, but less than 1% of the total radioactivity.

2.3.1.3 General Availability of Information. The main sources of data pertaining to waste generated at ANL-W from 1994 to 1999 are the ANL-W shipping records, IWTS, and waste characterization reports used for approval of waste receipts at the RWMC.

2.3.1.4 Data-Collection Approach. Data were collected by entering the ANL-W and IWTS data information into a spreadsheet and sorting the data by shipment year, destination, generating facility, and waste type to arrive at the total volumes of waste and total radioactivity from each facility, the waste-container types, and the waste types. Waste stream characteristics were gathered from ANL-W waste characterization reports and IWTS.

The estimated activity of unreported radionuclides was determined by multiplying appropriate scaling factors with Co-60, Cs-137, or Pu-239 activities that were reported. The shipping data, that

included the Co-60, Cs-137, or Pu-239 data (as well as some other radionuclides), were obtained from the IWTS database and were assumed to be correct.

2.3.1.5 Description of Waste Streams. The ANL-W waste sent to the SDA from 1994 to 1999 is divided into 13 waste streams (see Table 2-1). Some of these waste streams were extensions of the waste streams identified in previous reports: the Historical Data Task [HDT] (LMITCO 1995b) and the RPDT (LMITCO 1995a).

2.3.1.6 Asbestos Calculation. As noted previously in Sections 1 and 2.2, disposal data were collected from the RWMIS and IWTS databases. The RWMIS recorded approximate disposal of asbestos based on percent volume of the waste. The IWTS recorded approximate disposal of asbestos based on percent weight and percent volume of the waste, depending on the IWTS material profile. Therefore, the amount of asbestos was calculated in two different ways, depending on whether the amount of asbestos was recorded as percent weight or percent volume.

2.3.1.6.1 Percent Volume—The percent volume of asbestos is determined as follows:

$$\text{Best estimate for asbestos (g)} = [\text{percent volume of asbestos}] \times [\text{gross volume of container}] \times [\text{conversion of ft}^3 \text{ to m}^3] \times [\text{asbestos content}] \times [\text{conversion of lb to g}] \times [\text{density of asbestos}] \quad (1)$$

where

percent volume of asbestos	=	value recorded in the database for the container
gross volume of container	=	value recorded in database for the container
conversion of ft ³ to m ³	=	35.3 ft ³ /m ³
conversion of lb to g	=	454 g/lb
density of asbestos	=	16 lb/ft ³ (LMITCO 1995b, Table D-1)
asbestos content	=	0.15 (LMITCO 1995b, Table D-1).

The upper-bound value for asbestos was obtained by multiplying 1.5 times the same value for calculating the upper-bound value for asbestos in the HDT (LMITCO 1995b).

2.3.1.6.2 Percent Weight—The approximate percent weight of the asbestos was not recorded in the IWTS, but a range for the amount of asbestos in the container was provided. The mid-value range of the asbestos percent weight was used to ensure the best estimate:

$$\text{Best estimate for asbestos (g)} = [\text{percent weight of asbestos}] \times [\text{gross weight of container, (as recorded)}] \quad (2)$$

The upper value of the percent-weight range of the asbestos was used for the upper-bound value.

2.3.1.7 Scaling-Factor Analysis. The purpose of this section is to describe the methodology used to calculate the radioactive inventory of activation products, fission products, and TRU radioisotopes that probably were included but not reported in radioactive waste shipments made from ANL-W to the RWMC during the years 1994 to 1999. The information presented in this report updates the projected data that were discussed in the RPDT.

Table 2-1. Waste streams generated at Argonne National Laboratory-West from 1994 to 1999.

Waste Stream Number	Description of Waste
ANL-704-1	Contact-handled, nonprocessable LLW generated during manufacturing of metallic fuels and facility operations in the FMF.
ANL-752-1	Contact-handled, nonprocessable LLW generated during L&O facility operations, maintenance, modifications, and monitoring.
ANL-763-1	Contact-handled, LLW consisting of sludge solidified with grout, soil, rocks, and concrete pieces generated during the cleanup of the EBR-II leach pit.
ANL-765-1	Contact-handled, nonprocessable LLW generated during FCF operations, maintenance, modifications, and monitoring.
ANL-785-1	Remote-handled, subassembly LLW generated during nuclear fuel and materials experiments in the HFEF.
ANL-785-2	Contact-handled, nonprocessable LLW generated during HFEF operations, maintenance, modifications, and monitoring.
ANL-785-3	Remote-handled, LLW generated during HFEF operations, maintenance, modifications, and monitoring.
ANL-793-1	Contact-handled, nonprocessable LLW generated during SCMS, EBR-II, TREAT, and ZPPR facility operations, maintenance, modifications, and monitoring.
ANL-798-1	The remote-handled, nonprocessable LLW generated during facility maintenance, monitoring, and the evaporation of low-level liquid waste in shielded hot-air-drum evaporators.
ANL-ANL-1	Nonprocessed LLW consisting primarily of wood generated from maintenance and facility operations.
ANL-EBRI-2	NaK from EBR-I generated in 1955 and stored until treated in the mid-1990s at ANL-W.
ANL-INC-1	LLW waste generated from maintenance and facility operations. Waste was treated (incinerated) at Scientific Ecology Group.
ANL-TBS-1	LLW generated from a treatability study. Radioactive contaminated mercury light tubes crushed and mixed with chemically bonded phosphate ceramic technology for stabilization.

ANL-W = Argonne National Laboratory –West
 EBR-I = Experimental Breeder Reactor-I
 EBR-II = Experimental Breeder Reactor-II
 FCF = Fuel Cycle Facility
 FMF = Fuel Manufacturing Facility
 HFEF = Hot Fuel Examination Facility
 L&O = Laboratory and Office Building
 LLW = low-level waste
 SCMS = Sodium Components Maintenance Shop
 TREAT = Transient Reactor Test Facility
 ZPPR = Zero Power Physics Reactor

Unreported radionuclide activity was determined by multiplying isotope-dependent scaling factors with Co-60, Cs-137, or Pu-239 activities that were reported in the ANL-W waste shipments to the RWMC. The ANL-W shipping data (which included the activities of Co-60, Cs-137, and Pu-239, as well as other radionuclides) were obtained from the IWTS database.

Scaling factors used in the updated analysis, and those determined from CIDRA, are compared in Tables 2-2 through 2-5. The scaling factors shown in Tables 2-2 and 2-3 apply to activation products, and those shown in Tables 2-4 and 2-5 apply to fission products and actinides (including TRU isotopes). The difference between the corresponding table values (Table 2-2 versus Table 2-3, or Table 2-4 versus Table 2-5) is the amount of decay time that was assumed for the particular type of waste that was shipped. Most waste shipments used the data shown in Tables 2-3 or 2-4, and only a special shipment of NaK waste from EBR-I, processed in 1994, used the scaling factors shown in Table 2-5.

Table 2-2. A comparison of updated and CIDRA scaling factors for activation products assuming no decay time (0 years) prior to shipment to the Radioactive Waste Management Complex.

Primary Isotope	Isotope Half-Life (years)	Updated Best-Estimate Scaling Factors	Updated Upper-Bound Scaling Factors	CIDRA ^a Best-Estimate Scaling Factors	CIDRA ^a Upper-Bound Scaling Factors	Isotope Ratio
C-14	5.73E+03	7.00E-04	1.80E-03	N/A	N/A	C-14/Co-60
Na-22	2.61E+00	3.90E-02	5.91E-01	1.80E-07	8.90E-07	Na-22/Co-60
Cl-36	3.01E+05	2.55E-16	— ^b	N/A	N/A	Cl-36/Co-60
Cr-51	7.58E-02	1.06E+01	2.09E+01	3.90E-02	1.94E-01	Cr-51/Co-60
Mn-54	8.55E-01	1.69E+01	3.71E+01	1.05E-01	5.32E-01	Mn-54/Co-60
Fe-55	2.68E+00	N/A	N/A	N/A	N/A	—
Fe-59	1.22E-01	1.18E+00	1.46E+00	1.30E-07	6.60E+02	Fe-59/Co-60
Co-57	7.44E-01	1.03E+00	1.03E+00	1.94E-07	1.10E-06	Co-57/Co-60
Co-58	1.94E-01	7.77E+01	1.29E+02	1.56E-01	7.90E+01	Co-58/Co-60
Co-60	5.27E+00	1.00E+00	2.26E+00	1.00E+00	2.26E+06	Co-60/Co-60
Ni-59	7.60E+04	1.91E-03	7.40E-03	N/A	N/A	Ni-59/Co-60
Ni-63	1.00E+02	9.30E-02	5.54E-01	N/A	N/A	Ni-63/Co-60
Nb-94	2.00E+04	2.47E-05	5.90E-05	N/A	N/A	Nb-94/Co-60
Nb-95	9.58E-02	5.50E-03	1.25E-02	4.52E-08	1.94E-07	Nb-95/Co-60
Tc-99	2.13E+05	Tc-99/Cs-137 Use	Tc-99/Cs-137 Use	N/A	N/A	—

CIDRA = Contaminant Inventory Database for Risk Assessment

RPDT = Recent and Projected Data Task

SF_{be} = best-estimate scaling factor

a. The values shown were determined from the data reported in the Recent and Projected Data Task (RPDT) (LMITCO 1995a, Table 3-21b, pp. 3 to 63). For example, the Contaminant Inventory Database for Risk Assessment (CIDRA) SF_{be} for Nb-95 (relative to Co-60) was computed as follows: $4.52E-08 = 2.8E-02 / 6.2E+05 = \text{Nb-95 Ci/Co-60 Ci}$. The upper-bound CIDRA scaling factor for Nb-95 was determined as follows: $1.94E-07 = 1.2E-01 / 6.2E+05$. Note that the CIDRA scaling factors computed from data presented in RPDT Table 3-21b (1994 to 2003) produce different scaling factors than those obtained from data shown in RPDT, Table 3-20b (1984 to 1993).

b. The upper-bound scaling factor was not available for Cl-36. The best-estimate value for Cl-36 was reported for Cl-36 upper bound.

Table 2-3. A comparison of updated and CIDRA scaling factors for activation products assuming an average decay time of 0.5 years prior to shipment to the Radioactive Waste Management Complex.

Primary Isotope	Isotope Half-Life (years)	Updated ^b		CIDRA ^a		Isotope Ratio
		Best-Estimate Scaling Factors	Updated ^b Upper-Bound Scaling Factors	Best-Estimate Scaling Factors	CIDRA ^a Upper-Bound Scaling Factors	
C-14	5.73E+03	7.00E-03	1.90E-03	N/A	N/A	C-14/Co-60
Na-22	2.61E+00	3.60E-02	5.55E-01	1.80E-07	8.90 E-07	Na-22/Co-60
Cl-36	3.01E+05	2.72E-16	— ^c	N/A	N/A	Cl-36/Co-60
Cr-51	7.58E-02	1.17E-01	2.31E-01	3.90E-02	1.94E-01	Cr-51/Co-60
Mn-54	8.55E-01	1.21E+01	2.64E+01	1.05E	5.32E-01	Mn-51/Co-60
Fe-55	2.68E+00	N/A	N/A	N/A	N/A	—
Fe-59	1.22E-01	7.30E-02	9.10E-02	1.30E-02	6.60E-01	Fe-59/Co-60
Co-57	7.44E-01	6.93E-01	6.93E-01	1.94E-07	1.1E-06	Co-57/Co-60
Co-58	1.94E-01	1.39E+01	2.32E+01	1.56E-01	7.90E-01	Co-58/Co-60
Co-60	5.27E+00	1.00E+00	2.26E+00	1.00E+00	2.26E+00	Co-60/Co-60
Ni-59	7.60E+04	2.04E-03	7.90E-03	N/A	N/A	Ni-59/Co-60
Ni-63	1.00E+02	9.90E-02	5.89E-01	N/A	N/A	Ni-63/Co-60
Nb-94	2.00E+04	2.64E-05	6.30E-05	N/A	N/A	Nb-94/Co-60
Nb-95	9.58E-02	1.58E-04	3.57E-04	4.52E-08	1.94E-07	Nb-95/Co-60
Tc-99	2.13E+05	Tc-99/Cs-137 Use	Tc-99/Cs-137 Use	N/A	N/A	—

CIDRA = Contaminant Inventory Database for Risk Assessment

RPDT = Recent and Projected Data Task

SF_{be} = best-estimate scaling factor

a. The values shown were determined from the data reported in the Recent and Projected Data Task (RPDT), (LMITCO 1995a, Table 3-21b pp. 3 to 63). For example, the Contaminant Inventory Database for Risk Assessment (CIDRA) SF_{be} for Nb-95 (relative to Co-60) was computed as follows: $4.52\text{E-}08 = 2.8\text{E-}02 / 6.2\text{E+}05 = \text{Nb-95 Ci/Co-60 Ci}$. The upper-bound CIDRA scaling factor for Nb-95 was determined as follows: $1.94\text{E-}07 = 1.2\text{E-}01 / 6.2\text{E+}05$. Note that the CIDRA scaling factors computed from data presented in RPDT Table 3-21b (1994 to 2003) produce different scaling factors than those obtained from data shown in RPDT, Table 3-20b (1984 to 1993).

b. A decay time at Argonne National Laboratory-West of $t = 0.5$ years is assumed in the updated scaling factors.

c. The upper-bound scaling factor was not available for Cl-36. The best-estimate value for Cl-36 was reported for Cl-36 upper bound.

Table 2-4. A comparison of updated and CIDRA scaling factors for fission products and actinides assuming an average decay time of 0.5 years prior to shipment to the Radioactive Waste Management Complex. (The table values apply to all shipments except the NaK waste shipment).^a

Primary Isotope	Isotope Half-Life (years)	Updated ^b Best-Estimate Scaling Factors	Updated ^b Upper-Bound Scaling Factors	CIDRA ^a Best-Estimate Scaling Factors	CIDRA ^a Upper-Bound Scaling Factors	Isotope Ratio
H-3	1.23E+01	7.10E-03	2.19E+00	3.64E-01	2.36E+00	H-3/Cs-137
Sr-89	1.38E-01	5.86E-01	3.47E+00	4.91E-05	2.45E-04	Sr-89/Cs-137
Sr-90	2.90E+01	9.04E-01	5.34E+00	8.64E-01	4.27E+00	Sr-90/Cs-137
Y-90	7.30E-03	9.04E-01	5.34E+00	8.00E-01	3.82E+00	Y-90/Cs-137
Zr-95	1.75E-01	1.35E+00	6.73E+00	1.50E-02	9.10E-02	Zr-95/Cs-137
Tc-99	2.13E+05	1.36E-04	2.13E-04	N/A	N/A	Tc-99/Cs-137
Ru-106	1.02E+00	6.41E-01	3.30E+00	8.64E-04	6.00E-03	Ru-106/Cs-137
Ag-110m	6.84E-01	8.86E-05	4.47E-04	8.82E-05	4.54E-04	Ag-110m/Cs-137
Sn-113	3.15E-01	1.93E-03	9.50E-03	5.73E-03	2.82E-02	Sn-113/Cs-137
Sn-117m	3.72E-02	1.15E-10	5.73E-10	3.09E-03	1.55E-02	Sn-117m/Cs-137
Sb-124	1.65E-01	8.15E-05	4.08E-04	2.18E-04	1.09E-03	Sb-124/Cs-137
Sb-125	2.76E+00	6.70E-02	3.65E-01	5.00E-03	2.60E-02	Sb-125/Cs-137
Te-132	8.92E-03	0.00E+00	0.00E+00	N/A	N/A	Te-132/Cs-137
I-129	1.60E+07	2.79E-07	2.30E-03	N/A	N/A	I-129/Cs-137
I-131	2.20E-02	3.04E-07	1.46E-06	2.27E-03	1.09E-02	I-131/Cs-137
Cs-134	2.07E+00	7.50E-02	4.29E-01	3.82E-02	2.18E-01	Cs-134/Cs-137
Cs-137	3.02E+01	1.00E+00	1.53E+00	1.00E+00	2.18E-01	Cs-137/Cs-137
Ba-140	3.49E-02	3.32E-04	2.43E-03	1.36E-03	1.00E-02	Ba-140/Cs-137
La-140	4.60E-03	0.00E+00	0.00E+00	1.73E-03	1.27E-02	La-140/Cs-137
Ce-144	7.80E-01	4.74E+00	2.44E+01	1.18E-01	6.09E-01	Ce-144/Cs-137
Eu-152	1.34E+01	6.46E-05	9.63E-05	N/A	N/A	Eu-152/Cs-137
Eu-154	8.50E+00	4.00E-03	1.60E-01	3.00E-03	1.70E-02	Eu-154/Cs-137
Eu-155	4.73E+00	4.10E-02	2.59E-01	6.10E-02	3.82E-01	Eu-155/Cs-137
Ta-182	3.13E-01	3.00E-03	7.10E-03	9.0E-03	4.20E-02	Ta-182/Cs-137
Pb-210	2.23E+01	2.11E-13	3.21E-13	N/A	N/A	Pb-210/Cs-137
Ra-226	1.60E+03	5.18E-12	7.93E-12	N/A	N/A	Ra-226/Cs-137
Ra-228	5.76E+00	9.87E-17	1.51E-16	N/A	N/A	Ra-228/Cs-137
Ac-227	2.18E+01	2.82E-08	1.39E-07	N/A	N/A	Ac-227/Pu-239
Th-228	1.91E+00	4.63E-05	2.29E-04	N/A	N/A	Th-228/Pu-239
Th-229	7.30E+03	9.61E-09	4.74E-08	N/A	N/A	Th-229/Pu-239

Table 2-4. (continued).

Primary Isotope	Isotope Half-Life (years)	Updated ^b		CIDRA ^a		Isotope Ratio
		Best-Estimate Scaling Factors	Upper-Bound Scaling Factors	Best-Estimate Scaling Factors	Upper-Bound Scaling Factors	
Th-230	7.54E+04	5.61E-06	2.77E-05	N/A	N/A	Th-230/Pu-239
Th-232	1.40E+10	5.62E-13	2.77E-12	N/A	N/A	Th-232/Pu-239
Pa-231	3.28E+04	4.43E-07	2.19E-06	N/A	N/A	Pa-231/Pu-239
U-232	7.00E+01	1.31E-04	6.39E-04	N/A	N/A	U-232/Pu-239
U-233	1.59E+05	1.40E-05	6.90E-05	N/A	N/A	U-233/Pu-239
U-234	2.45E+05	1.00E+00	4.94E+00	1.00E+00	4.04E+00	U-234/U-234
U-235	7.04E+08	1.00E+00	4.94E+00	1.00E+00	2.50E+00	U-235/U-235
U-236	2.34E+07	1.00E+00	4.94E+00	N/A	N/A	U-236/U-236
U-238	4.47E+09	1.00E+00	4.94E+00	1.00E+00	1.79E+00	U-238/U-238
Np-237	2.14E+06	1.10E-03	5.0E-03	N/A	N/A	Np-237/Pu-239
Pu-238	8.77E+01	6.78E-01	3.32E+00	N/A	N/A	Pu-238/Pu-239
Pu-239	2.41E+04	1.00E+00	4.94E+00	1.00E+00	6.33E+00	Pu-239/Pu-239
Pu-240	6.56E+03	3.50E-02	1.44E+00	2.28E-03	1.19E-02	Pu-240/Pu-239
Pu-241	1.44E+01	1.22E-01	3.49E+02	N/A	N/A	Pu-241/Pu-239
Pu-242	3.76E+05	2.85E-08	1.41E-07	N/A	N/A	Pu-242/Pu-239
Am-241	4.32E+02	2.53E-04	1.95E+01	N/A	N/A	Am-241/Pu-239
Am-243	7.37E+03	5.98E-09	2.95E-08	N/A	N/A	Am-243/Pu-239
Cm-244	1.81E+01	2.22E-08	1.05E-07	N/A	N/A	Cm-244/Pu-239

CIDRA = Contaminant Inventory Database for Risk Assessment

a. The values shown were determined from the data reported in the Recent and Projected Data Task (LMITCO 1995a).

Table 2-5. A comparison of current and CIDRA scaling factors for fission products and actinides assuming an average decay time of 40 years prior to shipment to the Radioactive Waste Management Complex. (The table values apply to only the NaK waste shipment.)

Primary Isotope	Isotope Half-Life (years)	Updated ^b Best-Estimate Scaling Factors	Updated ^b Upper-Bound Scaling Factors	CIDRA ^a Best-Estimate Scaling Factors	CIDRA ^a Upper-Bound Scaling Factors	Isotope Ratio
H-3	1.23E+01	1.91E-03	5.87E-01	3.64E-01	2.36E+00	H-3/Cs-137
Sr-89	1.38E-01	0.00E+00	0.00E+00	4.91E-05	2.45E-04	Sr-89/Cs-137
Sr-90	2.90E+01	8.71E-01	5.15E+00	8.64E-01	4.27E+00	Sr-90/Cs-137
Y-90	7.30E-03	8.71E-01	5.15E+00	8.00E-01	3.82E+00	Y-90/Cs-137
Zr-95	1.75E-01	0.00E+00	0.00E+00	1.45E-02	9.09E-02	Zr-95/Cs-137
Tc-99	2.13E+05	3.36E-04	5.27E-04	N/A	N/A	Tc-99/Cs-137
Ru-106	1.02E+00	3.51E-12	1.81E-11	8.64E-04	6.18E-03	Ru-106/Cs-137
Ag-110m	6.84E-01	0.00E+00	0.00E+00	8.82E-05	4.45E-04	Ag-110m/Cs-137
Sn-113	3.15E-01	0.00E+00	0.00E+00	5.73E-03	2.82E-02	Sn-113/Cs-137
Sn-117m	3.72E-02	0.00E+00	0.00E+00	3.09E-03	1.55E-02	Sn-117m/Cs-137
Sb-124	1.65E-01	0.00E+00	0.00E+00	2.18E-04	1.09E-03	Sb-124/Cs-137
Sb-125	2.76E+00	8.14E-06	4.45E-05	4.82E-03	2.64E-02	Sb-125/Cs-137
Te-132	8.92E-03	0.00E+00	0.00E+00	N/A	N/A	Te-132/Cs-137
I-129	1.60E+07	6.90E-07	5.82E-03	N/A	N/A	I-129/Cs-137
I-131	2.20E-02	0.00E+00	0.00E+00	2.27E-03	1.09E-02	I-131/Cs-137
Cs-134	2.07E+00	3.24E-07	1.85E-06	3.82E-02	2.18E-01	Cs-134/Cs-137
Cs-137	3.02E+01	1.00E+00	1.53E+00	1.00E+00	2.18E-01	Cs-137/Cs-137
Ba-140	3.49E-02	0.00E+00	0.00E+00	1.36E-03	1.00E-02	Ba-140/Cs-137
La-140	4.60E-03	0.00E+00	0.00E+00	1.73E-03	1.27E-02	La-140/Cs-137
Ce-144	7.80E-01	6.70E-15	3.45E-14	1.18E-01	6.09E-01	Ce-144/Cs-137
Eu-152	1.34E+01	2.08E-05	3.09E-05	N/A	N/A	Eu-152/Cs-137
Eu-154	8.50E+00	4.00E-04	1.54E-02	3.27E-03	1.73E-02	Eu-154/Cs-137
Eu-155	4.73E+00	3.14E-04	1.97E-03	6.09E-02	3.82E-01	Eu-155/Cs-137
Ta-182	3.13E-01	0.00E+00	0.00E+00	9.08E-03	4.18E-02	Ta-182/Cs-137
Pb-210	2.23E+01	1.53E-13	2.33E-13	N/A	N/A	Pb-210/Cs-137
Ra-226	1.60E+03	1.26E-11	1.93E-11	N/A	N/A	Ra-226/Cs-137
Ra-228	5.76E+00	0.00E+00	0.00E+00	N/A	N/A	Ra-228/Cs-137
Ac-227	2.18E+01	8.03E-09	3.96E-08	N/A	N/A	Ac-227/Pu-239
Th-228	1.91E+00	2.82E-11	1.39E-10	N/A	N/A	Th-228/Pu-239
Th-229	7.30E+03	9.58E-09	4.73E-08	N/A	N/A	Th-229/Pu-239
Th-230	7.54E+04	5.61E-06	2.77E-05	N/A	N/A	Th-230/Pu-239
Th-232	1.40E+10	5.62E-13	2.78E-12	N/A	N/A	Th-232/Pu-239
Pa-231	3.28E+04	4.44E-07	2.19E-06	N/A	N/A	Pa-231/Pu-239
U-232	7.00E+01	8.85E-05	4.33E-04	N/A	N/A	U-232/Pu-239

Table 2-5. (continued).

Primary Isotope	Isotope Half-Life (years)	Updated ^b Best-Estimate Scaling Factors	Updated ^b Upper-Bound Scaling Factors	CIDRA ^a Best-Estimate Scaling Factors	CIDRA ^a Upper-Bound Scaling Factors	Isotope Ratio
U-233	1.59E+05	1.40E-05	6.90E-05	N/A	N/A	U-233/Pu-239
U-234	2.45E+05	1.00E+00	4.94E+00	1.00E+00	4.04E+00	U-234/U-234
U-235	7.04E+08	1.00E+00	4.94E+00	1.00E+00	2.50E+00	U-235/U-235
U-236	2.34E+07	1.00E+00	4.94E+00	N/A	N/A	U-236/U-236
U-238	4.47E+09	1.00E+00	4.94E+00	1.00E+00	1.79E+00	U-238/U-238
Np-237	2.14E+06	1.07E-03	5.26E-03	N/A	N/A	Np-237/Pu-239
Pu-238	8.77E+01	4.97E-01	2.43E+00	N/A	N/A	Pu-238/Pu-239
Pu-239	2.41E+04	1.00E+00	4.94E+00	1.00E+00	6.33E+00	Pu-239/Pu-239
Pu-240	6.56E+03	3.54E-02	1.43E+00	2.28E-03	1.19E-02	Pu-240/Pu-239
Pu-241	1.44E+01	1.81E-02	5.18E+01	N/A	N/A	Pu-241/Pu-239
Pu-242	3.76E+05	2.85E-08	1.41E-07	N/A	N/A	Pu-242/Pu-239
Am-241	4.32E+02	2.38E-04	1.83E+01	N/A	N/A	Am-241/Pu-239
Am-243	7.37E+03	5.96E-09	2.94E-08	N/A	N/A	Am-243/Pu-239
Cm-244	1.81E+01	4.90E-09	2.33E-08	N/A	N/A	Cm-244/Pu-239

CIDRA = Contaminant Inventory Database for Risk Assessment

a. The values shown were determined from the data reported in the Recent and Projected Data Task (LMITCO 1995a).

All scaling factors are dependent on the amount of waste holdup or decay time while the waste is stored or processed at ANL-W (i.e., the time prior to shipment to the RWMC). The assumed amount of decay time for all waste shipments, except for the NaK shipments from EBR-I operations, was 6 months ($t = 0.5$ years). The holdup time for activated metals (e.g., subassembly hardware) and LLW was estimated from information provided by B. R. Adams and R. P. Grant at ANL-W. Grant said that ANL-W assumes a 3-year holdup time for all contact-handled LLW.^a Adams researched 27 shipments of subassembly hardware made between 1979 and 1991.^b Based on these shipments, as shown in Table 2-6, the average amount of holdup time for subassembly hardware at ANL-W prior to shipment to the RWMC, was 602 days (1.65 years), and the minimum holdup time was 52 days. A holdup time of 6 months was selected for all waste (except the NaK waste shipments) for the scaling-factor analysis presented here. This assumption provides a sufficient amount of time for many short-lived radionuclides to decay completely, while this assumption has little effect on the calculated inventory of the long-lived radionuclide inventory.

A decay time of 40 years was assumed for the NaK (i.e., a coolant used at EBR-I) waste shipments in contrast to the short holdup time for most ANL-W waste shipments. Because only fission products are the key constituents present in the NaK, no activation-product scaling factors were determined for the NaK waste shipments.

a. Grant, Roy P., Argonne National Laboratory-West, Personal Communication with Bruce R. Adams, Argonne National Laboratory-West, September 2000.

b. Adams, B., Interoffice e-mail to M. L. Carboneau, September 20, 2000, "Holdup Time for Subassembly Hardware at ANL-W," Idaho National Engineering and Environmental Laboratory, Bechtel BWXT Idaho, LLC, Idaho Falls, Idaho.

Table 2-6. Average waste storage and processing decay time at Argonne National Laboratory-West prior to shipment to the Radioactive Waste Management Complex.

Date of Hardware Removal from the Reactor Core	Date Waste Was Placed in the Shipping Can	Holdup Time (days)
12/05/79	10/05/87	2,861
10/04/81	8/11/86	1,772
6/01/83	8/15/86	1,171
12/09/83	9/29/87	1,390
4/09/84	11/21/86	956
4/16/84	11/21/86	949
4/07/85	5/29/85	52
7/23/85	2/19/88	941
8/26/85	1/29/86	156
8/26/85	11/21/86	452
8/26/85	5/04/87	616
2/02/86	11/21/86	292
3/27/86	11/21/86	239
11/19/86	2/19/88	457
12/5/86	11/06/87	336
12/10/86	2/05/88	422
12/12/86	5/04/87	143
1/29/87	9/29/87	243
2/10/87	5/04/87	83
2/10/87	5/04/87	83
8/24/87	2/04/88	164
8/24/87	3/07/88	196
11/02/87	2/05/88	95
11/02/87	2/18/91	1,204
11/01/88	7/18/89	259
5/25/90	2/11/91	262
3/29/91	7/14/92	473
—	Minimum holdup	52
—	Maximum holdup	2,861
—	Mean value	602

Time-dependent scaling factors (SF[t]s) can be computed from the initial (t = 0) scaling factors (SF[0]s). The following equation is used to determine the scaling factor for Mn-54 relative to Co-60 at some future time (t), where the initial scaling factor (SF[0]) is known at t = 0:

$$SF(0) = \text{initial activity of Mn-54 [Mn-54 (t = 0)] divided by the initial activity of Co-60 [Co-60(t = 0)]}. \quad (3)$$

The SF(t) for Mn-54 is then computed as follows:

$$SF(t) = \text{final activity of Mn-54 [Mn-54(t)] divided by the final Co-60 activity [Co-60(t)]}. \quad (4)$$

Therefore,

$$SF(t) = \text{Mn-54(t) / Co-60(t)} = [\text{Mn-54(0)} \times \exp(-\lambda_1 t)] / [\text{Co-60(0)} \times \exp(-\lambda_2 t)] = SF(0) \times [\exp(-\lambda_1 t) / \exp(-\lambda_2 t)] \quad (5)$$

where

$$\lambda_1 = \text{decay constant for Mn-54 (key isotope)} = \ln(2) / 0.855 \text{ year} = 0.8107 \text{ per year}$$

$$\lambda_2 = \text{decay constant for Co-60 (reference isotope)} = \ln(2) / 5.27 \text{ year} = 0.1315 \text{ per year.}$$

For Mn-54:

$$SF(t) = SF(0) \times [\exp(-0.8107 \times 0.5) / \exp(-0.1315 \times 0.5)] = SF(0) \times 0.712. \quad (6)$$

From Table 2-2, the Mn-54 best-estimate scaling factor (SF_{be}) at t = 0 is 16.945. Therefore, the SF_{be} for Mn-54 at t = 0.5 years is computed as follows:

$$SF(0) \times 0.712 = 16.945 \times 0.712 = 12.064 \text{ (which is the value shown in Table 2-2)}. \quad (7)$$

Therefore,

$$SF(t) = SF(0) \times [\exp(-\lambda_1 t) / \exp(-\lambda_2 t)] . \quad (8)$$

The ANL-W scaling factors at t = 0 (e.g., SF[0]) were determined from reactor physics calculations computed shortly after a reactor scram (i.e., an emergency shutdown of a reactor). K. Bunde (ANL-W) and R. McKnight (Argonne National Laboratory-East [ANL-E]) provided the calculated inventory data (McKnight 2000a).^c If the amount of time (t) the specific radioactive material were stored at ANL-W prior to shipment to the RWMC is known, then the scaling factor at the time of shipping (SF[t]) can be computed using Equation (1) for each radioisotope of interest. Similar relationships were established for scaling factors based on Cs-137 and Pu-239. Scaling factors based on Co-60 were used to estimate the shipped inventories of certain radionuclides (i.e., those listed in 10 CFR 61) (e.g., C-14, Ni-59, Ni-63, and Nb-94) as well as some other radionuclides shown in Table 2-3. Scaling factors based on Cs-137 were used to estimate the as-shipped inventories of several fission products, in particular Sr-90, Tc-99, I-129, and several other radionuclides shown in Table 2-4. The scaling factors based on Pu-239 were used to estimate the shipped activities of several important actinides and TRU waste (e.g., isotopes of thorium, protactinium, neptunium, plutonium, americium, and curium), and also are shown in Table 2-4.

2.3.1.7.1 Best-Estimate Scaling Factors—The SF_{bes} for the current study were usually determined from reactor physics (ORIGEN) calculations performed by K. Bunde (ANL-W) or

c. McKnight, R. D., Interoffice e-mail to M. L. Carboneau, August 7, 2000a, Idaho National Engineering and Environmental Laboratory, Bechtel BWXT Idaho, LLC, Idaho Falls, Idaho.

R. McKnight (ANL-E) and transmitted to the INEEL (McKnight 2000a, b)^d The Bunde and McKnight calculations produced a detailed radionuclide inventory of activation products (from subassembly hardware), as well as fission products and TRU isotopes produced in the EBR-II fuel pins. These results were calculated at or near reactor scram ($t = 0$). Each calculation considered the irradiation of either a Mark-I, Mark II, or Mark III driver subassembly within the EBR-II reactor under typical operating conditions. Bunde modeled both Mark-I and Mark-II subassemblies in several reactor locations, and McKnight modeled a single Mark-III driver subassembly. Both Mark-II and Mark-III driver subassemblies are similar except for the number of fuel elements per subassembly (i.e., 91 per Mark-II and 61 per Mark-III) and the maximum burnup condition for each subassembly.

The SF_{be} were then computed as the average of several separately determined scaling factors. As shown below, one scaling factor was based on ANL-W data and another based on ANL-E data.

$$SF(0) = [SF_1 + SF_2] / 2 \quad (9)$$

where

SF_1 = the ratio of the curie inventory of the key (or target) isotope at $t = 0$ divided by the curie inventory of the reference isotope at $t = 0$ using the ANL-W results (set #1) (based on Bunde calculations)^e

SF_2 = the scaling factor based on the ANL-E results (set #2) (see footnote c, p. 17, and footnote d, p. 20).

Averaging was done whenever information was available from multiple sources. However, when only one source was available, the SF_{be} was determined from the one available data set (i.e., $SF[0] = SF_1$ or $SF[0] = SF_2$). The Bunde calculations (see footnote e below) usually provided information pertinent to activation products generated in neutron-activated stainless steel. The McKnight calculations generally provided information on fission product and TRU inventories found in the fuel, however, McKnight calculated some activation product inventories for the stainless steel fuel pins. In some special cases (e.g., Cl-36 was produced from impurities contained in stainless steel), no information was available from Bunde or McKnight. In this case, an ORIGEN2 calculation was performed. Again, the ORIGEN2 calculation modeled the irradiation of an EBR-II subassembly under typical operating conditions. The INEEL-calculated results were then used to determine the inventories of Cl-36 and Co-60 at reactor scram ($t = 0$) contained in stainless steel. This special ORIGEN2 calculation also was used to double check the results reported by Bunde and McKnight. In any case, the SF_{bes} ($SF_u[0]$) at $t = 0$ were determined as the ratio of the target inventory to the reference isotope inventory, with the denominator being the inventory of the reference isotope (i.e., Co-60, Cs-137, or Pu-239).

2.3.1.7.2 Upper-Bound Scaling Factors—Upper-bound scaling factors were determined in addition to SF_{be} (e.g., $SF_{be}[0]$ and $SF_{be}[t]$). The purpose of the SF^{up} was to estimate the maximum inventories of several key isotopes not reported in the original shipping manifests, but expected to be present in the waste. Estimating the maximum inventories was accomplished by multiplying the SF^{up} by the reported activity of the reference isotope.

d. McKnight, R. D., Interoffice e-mail to M. L. Carboneau, August 7, 2000b, Idaho National Engineering and Environmental Laboratory, Bechtel BWXT Idaho, LLC, Idaho Falls, Idaho.

e. Bunde, K., Interoffice e-mail to M. L. Carboneau, April 7, 2000, "ANL-W Curies Inventory in the EBR-II Subassemblies," Idaho National Engineering and Environmental Laboratory, Bechtel BWXT LLC, Idaho, Idaho Falls, Idaho.

Unlike the methodology used to generate the SF_{be} (as discussed above), no single method provided enough information to determine a reasonable SF^{up} for every possible isotope or situation. Many different sources of information were relied on to determine the SF^{up} , as shown in Tables 2-3, 2-4, and 2-5. When sufficient information was available to determine an SF_{be} , based on at least two data sets, then a standard deviation value could be determined and the SF^{up} was defined as follows:

$$SF^{up}(0) = SF_{be}(0) + 2 \times \sigma = \mu + 2\sigma \quad (10)$$

where

SF^{up} = upper-bound scaling factor

$SF_{be}(0)$ = best-estimate scaling factor $t = 0$

σ = standard deviation.

For example, if two data sets were available, then the SF_{be} and SF^{up} were computed as follows:

$$SF_{be}(0) = \mu = [SF_1 + SF_2] / 2; \sigma^2 = [(SF_1 - \mu)^2 + (SF_2 - \mu)^2], \text{ and } SF^{up}(0) = \mu + 2\sigma. \quad (11)$$

2.3.1.7.3 Special Cases—The above procedure did not provide reasonable information (e.g., when $\sigma = 0$) to determine a maximum scaling factor for some isotopes. Alternative techniques were employed in such cases to estimate the SF^{up} . For example, the SF^{up} for Sr-89 was determined from the Sr-90 data as the ratio of the SF^{up} to the SF_{be} for Sr-90, multiplied by the SF_{be} for Sr-89. That is,

$$SF^{up}(Sr-89) = [SF^{up}(Sr-90) / SF_{be}(Sr-90)] \times SF_{be}(Sr-89). \quad (12)$$

Because Y-90 has a short half-life, the assumption was made that Y-90 would always be in secular equilibrium with its parent isotope, Sr-90. Therefore,

$$SF_{be}(Y-90) = SF_{be}(Sr-90) \text{ and } SF^{up}(Y-90) = SF^{up}(Sr-90). \quad (13)$$

The SF^{up} s for the current analysis were estimated from the RPDT (e.g., computed from inventory data shown in the RPDT [LMITCO 1995a, Table 3-21b]) in many other cases (e.g., for Zr-95, Ru-106, Ag-110m, Sn-113, Sn-117m, Sb-124, Sb-125, I-131, Cs-134, Cs-137, Ba-140, La-140, Ce-144, and Eu-144). For instance, the SF^{up} for Cs-134 was determined as follows:

$$SF^{up}(Cs-134) = (CIDRA SF^{up} \text{ from Table 2-4}) / (CIDRA SF_{be} \text{ from Table 2-3}) \times (\text{current } SF_{be} \text{ for Cs-134 from Table 2-4}) = (0.218/0.03818) \times 0.075 = 0.428. \quad (14)$$

Equivalently,

$$SF^{up}(Cs-134) = (CIDRA SF^{up} \text{ from the RPDT [LMITCO 1995a, Table 3-21b]}) / (CIDRA \text{ best-estimate inventory shown in the RPDT [LMITCO 1995a, Table 3-21b]}) \times (\text{current } SF_{be} \text{ for Cs-134 from Table 2-4}) = (2.4 \text{ Ci}/0.42 \text{ Ci}) \times 0.075 = 0.428. \quad (15)$$

2.3.1.8 Estimated Inventories of Radioisotopes in Argonne National Laboratory-West Waste Shipments. After SF_{be} and SF^{up} were determined, calculated curie inventories of radioisotopes probably present, but not reported, in the ANL-W waste shipments to the RWMC from 1993 to 1999 could be determined. When a radioisotope was listed in the IWTS database, no scaling-factor analysis was necessary and the reported data were accepted as listed. However, when a key isotope was not reported in the waste shipment records, then the activity of this isotope could be determined by

multiplying the appropriate scaling factor (e.g., SF_{be} or SF^{up} obtained from Tables 2-3 through 2-5) by its reference isotope activity (Co-60, Cs-137, or Pu-239). The results of these calculations are shown in subsequent sections.

2.3.2 Test Reactor Area

2.3.2.1 Waste Generator. The TRA is located approximately 8 km (5 mi) north of CFA and approximately 3 km (2 mi) west of the INTEC at the INEEL site.

The major operating facility at TRA is the Advanced Test Reactor (ATR), which is located in TRA-670 and has been in operation since 1967. Some of the support facilities for the ATR include the TRA hot cells (TRA-632), the Radiation Measurements Laboratory (TRA-661), the Nuclear Materials Inspection Storage Facility (TRA-621), and the Hydraulics Test Facility (TRA-666). Other major facilities that were important in the past but now have been deactivated are the Materials Test Facility (formerly called the Materials Test Reactor Facility) (ATR-603), shut down in 1970, and the Gamma Facility (TRA-641), and the Engineering Test Reactor (TRA-642), which both were closed in 1981. The Engineering Test Reactor Critical (TRA-654), closed in 1980, and the Advanced Reactivity Measurements Facility (TRA-660), closed in 1992, both may be reactivated at some point in the future. The designation recognized under the FFA/CO and CERCLA for TRA is WAG 2 (see Figure 1-1).

The ATR uses highly enriched uranium (i.e., 93% U-235 by mass) as its nuclear fuel. The fuel is contained in fuel-element assemblies composed of multiple fuel plates. The central core of each fuel plate contains a matrix of uranium and aluminum called UAL, and is covered by an outer layer of pure aluminum. The reactor core is cooled and neutron-moderated with water. The ATR has a beryllium reflector that surrounds the reactor core. The beryllium reflector and core internals have required changing out every 8 or 9 years. These beryllium reflectors previously had been disposed of at the RWMC; however, from 1994 to 1999, no shipments of beryllium were made.

2.3.2.2 Generation of the Waste. The waste shipped from TRA to the RWMC comes primarily from the operation of the ATR and examination of irradiated experimental assemblies in the TRA hot cells. This radioactive waste contains radioactive fission products produced in the nuclear fuel and radioisotopes produced by neutron activation. The nuclear-fuel-produced radioactivity is typically classified as mixed fission products; however, some activation products are associated with certain fuels. Neutron activation products are typically classified as mixed activation products. The actual distribution of specific nuclides in either mixed fission products or mixed activation products depends on the reactor fuel and the process that generated the waste.

The ATR began full-power operation in 1967. Its core has a four-leaf clover shape and there are nine major regions for experiments. The power for each region can be tailored to meet experimenter requirements. The maximum power level of the ATR is 250 MW; however, it typically operates at a power level of about 125 MW. The amount of core loading material for the ATR is approximately 40 kg of U-235. A canal is used to store irradiated and unirradiated fuel and irradiated experiment assemblies. Irradiated fuel is stored temporarily in the facility canal for a cooling period and then shipped to INTEC for longer-term storage.

Activation products are produced when neutrons are captured or otherwise interact to produce radionuclides. Neutron interactions can occur in the reactor fuel, causing activated radionuclides to be carried along with the fission products. Neutrons also can interact with reactor and experimental structural components, resulting in fixed radionuclide contamination in those components. Through corrosion, this fixed contamination can enter the reactor coolant. Once in the coolant system, the radionuclides and the fission products can potentially contaminate the same items. Therefore, radioactive waste generated by

test-reactor operations and support activities is a mixture of fission products and activation products. In addition to fission products and activation products, TRU radionuclides are produced in a reactor.

Since ATR began operation, core internal changeouts were performed in 1977, 1986, and 1994. Some irradiated and contaminated core parts from the latest core internal changeout were sent to the RWMC in 1994 and 1995. These contributed by far the largest portion of radioactivity from TRA. Another significant contributor was a number of lithium targets that were irradiated within the ATR core. In addition, during ATR operation, radionuclides that become entrained in the coolant loop water are scrubbed out in resin beds. These resin beds are changed out periodically and put in containers that are shipped to the RWMC. The resins also make a significant contribution to the total activity in the waste sent to the RWMC. The hot cells also generate waste at TRA. Almost all items removed from the hot cells are considered to be radioactive. If these items are no longer required, they are classified as radioactive waste.

No nonradiological contaminants were sent to the RWMC from TRA during the period from 1994 to 1999. General categories of waste sent to RWMC from TRA from 1994 to 1999 are as follows:

- Ion-exchange resins used to clean the reactor coolant water
- Dry, noncompactible waste such as wood, metal, glass, and concrete
- Parts from reactor core changeouts
- Waste from hot-cell activities
- Contaminated sludge from the resin beds
- Miscellaneous (e.g., lithium New Production Reactor [NPR] targets and treatability study monoliths [Portland cement]).

2.3.2.3 General Availability of Information. The main source of data pertaining to TRA waste shipments to the RWMC is the IWTS database. Additional information has been obtained from various technical reports and letters. The information from these reports and letters has been used primarily to assess the accuracy of IWTS and to evaluate scaling factors for use in updating the radionuclide inventory of waste shipments sent to the RWMC. Letter reports of resin-radionuclide analysis from the TRA Radiation Measurements Laboratory (between 1997 and 1999) were used to update scaling factors for resins.

2.3.2.4 Data-Collection Approach. Integrated Waste Tracking System entries for waste volume, mass, and radioactivity levels were used to collect data for TRA through the timeframe from 1994 to 1999. The data were supplemented with various reports, letters, and documents referenced in this supplement. Contacts also were made with TRA personnel to clarify and refine data found in the reference materials (see Section 2.3.2.6 below).

2.3.2.5 Description of Waste Streams. The TRA waste for 1994 to 1999 was divided into the seven general categories listed in Table 2-7. Resins and noncompactible waste composed the bulk of the overall weight and volume, while the core parts and NPR targets contributed most of the activity (see Table 2-29).

Table 2-7. Waste streams sent to the Radioactive Waste Management Complex from 1994 to 1999.

Waste Stream Numbers	Description
TRA-603-1	Resins
TRA-603-27	Noncompactible waste (i.e., metals, wood, and glass)
TRA-603-4	Core and loop components (from 1994 changeout)
TRA-632-2	Hot cell remote-handled waste
TRA-603-35	NPR-irradiated target material
TRA-603-6	TRA-645 cold well sludge
TRA-632-3	Portland cemented waste (treatability study monoliths)

NPR = New Production Reactor
TRA = Test Reactor Area

2.3.2.6 Scaling-Factor Analysis

2.3.2.6.1 Resins—The scaling factors for resins in the RPDT, for the period 1984 to 1993, were used as a starting point. However, more recent scaling factors have been developed.^f A number of sources spanning the years 1985 through 1999 were referenced. Three sources (Best et al. 1985; Best and Miller 1987; Vance 1988) are EPRI documents that include resin-nuclide data for pressurized water reactors (PWRs) and boiling water reactors. Because the PWR is the type of reactor most closely related to the ATR, only PWR resin data were considered in this evaluation. Though EPRI data must be used with caution because some hardware and operational differences exist between a standard PWR and the ATR, in a very general sense, the EPRI data sets agree with data from TRA. The following references were used to develop scaling factors for resins:

- *Radionuclide Correlations in Low-Level Radwaste* (Best et al. 1985)
- *Updated Scaling Factors in Low-Level Radwaste* (Best and Miller 1987)
- *Assessing the Impact of NRC Regulation 10 CFR 61 on the Nuclear Industry* (Vance 1988)
- *TRA Activity Weighting Factors/Physical and Chemical Properties of C-14, Tc-99 and I-129* (Akers 1994)
- *Scaling Factors for Waste Activities Measured by G-M Method* (Harker 1995)
- *A Comprehensive Inventory of Radiological and Nonradiological Contaminants in Waste Buried or Projected to Be Buried in the Subsurface Disposal Area of the INEL RWMC During the Years 1984–2003* (LMITCO 1995a)

f. Cotton, G. B., Interdepartmental Communication, September 2, 1999, “Estimate of Select Radionuclides Curie Content Generated by TRA Reactors and Disposed at the RWMC in Resins,” GBC-01-00, Idaho National Engineering and Environmental Laboratory, Lockheed Martin Idaho Technologies Company, Idaho Falls, Idaho.

- *Radioactive Waste Characterization Requirements for Reactor-Generated Low-Level Waste* (Tyger 1999)
- J. A. Logan letter to T. L. Clements, Jr.^g

Most of the time, the resin activity data contained in these reports are listed as relative activities for the various nuclides. The data have been converted to ratios relative to Co-60, Cs-137, and Pu-239. The scaling factors for Co-60 are appropriate for its activation products (i.e., H-3, C-14, Fe-55, Ni-59, Ni-63, and Nb-94). The Cs-137 and Pu-239 scaling factors are appropriate for fission products (i.e., Tc-99, Cs-137, Ce-144, Eu-154, and Eu-155) and for fuel and fuel activation products (i.e., U-234, U-235, U-236, Np-237, Pu-238, Pu-239, Pu-240, Pu-241, Am-241, Cm-242, and Cm-244).

The appropriate scaling factors based on C-60, Cs-137, and Pu-239 are listed in Table 2-8. The scaling factor for Nb-94 originated from an interoffice correspondence (see footnote f, p. 24). The scaling factor for I-129, originating from an engineering analysis (Harker 1995), is used as a more conservative estimate.

More than 40 reports, in the form of interdepartmental communications from the TRA Radiation Measurements Laboratory, were used to create a radioactive nuclide activity database for the resins. The reports span the time period from the end of 1996 through mid-1999 and include not only radionuclide activities but estimated measurement uncertainties. The data were used to calculate new resin scaling factors for specific nuclides where enough data were available (typically six to 12 data points), in a manner similar to that described by Abbott.^h The data could be naturally separated into the following four groups: (1) M-18 and M-19 anion resin beds, (2) M-16 and M-17 cation resin beds, (3) TRA-605 Warm Waste Treatment Facility resin beds, and (4) TRA-605 Warm Waste Treatment Facility mixed beds. Results showed that for certain radionuclides, the scaling factors for the anion beds were much different than for the other categories. Two separate scaling factors were calculated for the resins for these radionuclides, one for anion beds and one for the cation and mixed resin beds. Some radionuclide scaling factors revealed no difference between the anion or cation resin beds; therefore, only a single scaling factor was calculated. The updated scaling factors have been incorporated into Table 2-8.

The data on TRA resins from the TRA Radiation Measurements Laboratory also included estimated measurement uncertainties. Using standard statistical analysis, the measurement uncertainties were used to calculate standard deviations for each of the scaling factors. In Table 2-9, the RSD listed in Column 3 is defined as the standard deviation divided by the mean. The RSDs listed as “0,” “1,” or “5” are the recommended values listed in the RPDT, (LMITCO 1995a, Section 5.4.3). Details about the development of these RSDs also can be found in Einerson and Smith (1995). The SF^{up}s are calculated for RSDs “0,” “1,” and “5” by adding the (mean) scaling factor to the “RSD × scaling factor,” or simply by multiplying the scaling factor by one plus the RSD. For the rest of the RSDs calculated from uncertainty data, a two-standard-deviation approach is taken (e.g., the RSD is doubled in the calculations described above).

g. Logan, J. A., Interdepartmental Communication to T. L. Clements, Jr., September 9, 1999, “Assessment of Neutron-Activation Products in Low-Level Waste Discharged from Nuclear Reactors at the Test Reactor Area and Sent to the Radioactive Waste Management Complex for Disposal,” Attachment D, “Best-Estimate Radionuclide Inventories for Advanced Test Reactor Components,” JAL-04-99, Idaho National Engineering and Environmental Laboratory, Lockheed Martin Idaho Technologies Company, LLC, Idaho Falls, Idaho.

h. Abbott, M. L., Interdepartmental Communication to J. A. Logan, April 14, 1998, “Estimated C-14 Inventory in TRA Resin Shipments to the RWMC,” MLA-03-98, Idaho National Engineering Laboratory, Lockheed Martin Idaho Technologies Company, Idaho Falls, Idaho.

Table 2-8. Resin scaling factors from the Recent and Projected Data Task (LMITCO 1995a).

Nuclide Scaling Ratio	Scaling Factor	Scaling Factor Reference
H-3/Co-60	7.35E-04	RPDT (LMITCO 1995a, Table 7, pp. 2 to 31)
C-14/Co-60	6.32E-03	Best and Miller (1987)
Fe-55/Co-60	2.94E-01	Best and Miller (1987)
Ni-59/Co-60	4.12E-03	Best and Miller (1987)
Ni-63/Co-60	4.12E-01	Best and Miller (1987)
C-60/Co-60	1.00E+00	Best and Miller (1987)
Sr-90/Co-60	4.12E-01	Best and Miller (1987)
Nb-94/Co-60	9.39E-04	Cotton (1999) ^a
Tc-99/Cs-137	4.84E-05	RPDT (LMITCO 1995a, Table 7, pp. 2 to 31)
I-129/Cs-137	1.00E-04	Harker (1995)
Cs-137/Cs-137	1.00E+00	RPDT (LMITCO 1995a, Table 7, pp. 2 to 31)
Ce-144/Cs-137	2.16E-02	RPDT (LMITCO 1995a, Table 7, pp. 2 to 31)
Eu-154/Cs-137	2.35E-02	RPDT (LMITCO 1995a, Table 7, pp. 2 to 31)
Eu-155/Cs-137	1.00E-02	RPDT (LMITCO 1995a, Table 7, pp. 2 to 31)
U-234/Pu-239	9.13E-02	Croff (1980) and Schnitzler (1994) ^b
U-235/Pu-239	2.00E-03	Croff (1980) and Schnitzler (1994) ^b
U-236/Pu-239	3.48E-02	Croff (1980) and Schnitzler (1994) ^b
Np-237/Pu-239	5.65E-02	Croff (1980) and Schnitzler (1994) ^b
Pu-238/Pu-239	3.91E+00	RPDT (LMITCO 1995a, Table 7, pp. 2 to 31)
Pu-239/Pu-239	1.00E+00	RPDT (LMITCO 1995a, Table 7, pp. 2 to 31)
Pu-239/Co-60	6.76E-05	RPDT (LMITCO 1995a, Table 7, pp. 2 to 31)
Pu-240/Pu-239	5.43E-01	Croff (1980) and Schnitzler (1994) ^b
Pu-241/Pu-239	3.26E+02	RPDT (LMITCO 1995a, Table 7, pp. 2 to 31)
Am-241/Pu-239	9.13E+01	RPDT (LMITCO 1995a, Table 7, pp. 2 to 31)
Cm-242/Pu-239	6.09E+00	RPDT (LMITCO 1995a, Table 7, pp. 2 to 31)
Cm-244/Pu-239	2.83E+00	RPDT (LMITCO 1995a, Table 7, pp. 2 to 31)

RPDT = recent and project data task

a. Cotton, G. B., Interdepartmental Communication, September 2, 1999, "Estimate of Select Radionuclides Curie Content Generated by TRA Reactors and Disposed at the RWMC in Resins," GBC-01-00, Idaho National Engineering and Environmental Laboratory, Lockheed Martin Idaho Technologies Company, Idaho Falls, Idaho.

b. Schnitzler, G. B., Interdepartmental Correspondence to E. B. Nieschmidt, February 10, 1994, "Radioisotopes in ATR Fuel Elements," BGS-2-94, Idaho National Engineering and Environmental Laboratory, EG&G Idaho, Idaho Falls, Idaho.

Table 2-9. Updated resin scaling factors.

Nuclide Scaling Ratio	Scaling Factor	Relative Standard Deviation ^a	Upper-Bound Scaling Factor	Scaling Factor
H-3/Co-60	1.91E-01	0.25	2.87E-01	TRA anion beds
	1.74E-03	0.58	3.76E-03	TRA cation and mixed beds
C-14/Co-60	1.13E+00	0.23	1.65E+00	TRA anion beds
	3.92E-04	0.06	4.39E-04	TRA cation and mixed beds
Cr-51/Co-60	1.10E+03	0.23	1.61E+03	TRA anion beds
	5.29E-01	0.23	7.72E-01	TRA cation and mixed beds
Mn-54/Co-60	3.28E-02	0.13	4.13E-02	TRA all beds
Fe-55/Co-60	1.37E+00	0.30	2.19E+00	TRA anion beds
	6.75E-02	0.26	1.03E-01	TRA cation and mixed beds
Ni-59/Co-60	4.12E-03	5	2.47E-02	RPDT (LMITCO 1995a, Table 7, pp. 2 to 31)
Ni-63/Co-60	1.02E+00	0.27	1.57E+00	TRA anion beds
	6.05E-02	0.06	6.78E-02	TRA cation and mixed beds
Co-58/Co-60	4.50E-02	0.4	8.10E-02	TRA all beds
Co-60/Co-60	1.00E+00	1	2.00E+00	Not available
Sr-90/Co-60	7.18E-03	0.12	8.90E-03	TRA all beds
Nb-94/Co-60	9.39E-04	5	5.63E-03	Cotton (1999) ^b
Tc-99/Cs-137	4.84E-05	5	2.90E-04	RPDT (LMITCO 1995a, Table 7, pp. 2 to 31)
I-129/Cs-137	1.00E-04	5	6.00E-04	Harker (1995, Table 3)
Cs-137/Cs-137	1.00E+00	1	2.00E+00	Not available
Cs-137/Co-60	4.56E-01	1	9.12E-01	TRA all beds
Ce-144/Cs-137	2.16E-02	5	1.30E-01	RPDT (LMITCO 1995a)
Eu-154/Cs-137	2.35E-02	5	1.41E-01	RPDT (LMITCO 1995a)
Eu-155/Cs-137	1.00E-02	5	6.00E-02	RPDT (LMITCO 1995a)
U-234/Pu-239	9.13E-02	0	9.13E-02	Croff (1980) and Schnitzler (1994) ^c
U-235/Pu-239	2.00E-03	0	2.00E-03	Croff (1980) and Schnitzler (1994) ^c
U-236/Pu-239	3.48E-02	0	3.48E-02	Croff (1980) and Schnitzler (1994) ^c
Np-237/Pu-239	5.65E-02	5	3.39E-01	Croff (1980) and Schnitzler (1994) ^c
Pu-238/Pu-239	3.91E+00	5	2.35E+01	RPDT (LMITCO 1995a, Table 7, pp. 2 to 31)
Pu-239/Pu-239	1.00E+00	5	6.00E+00	Not available
Pu-239/Co-60	6.76E-05	5	4.06E-04	RPDT (LMITCO 1995a, Table 7, pp. 2 to 31)
Pu-240/Pu-239	5.43E-01	5	3.26E+00	Croff (1980) and Schnitzler (1994) ^c
Pu-241/Pu-239	3.26E+02	5	1.96E+03	RPDT (LMITCO 1995a)
Am-241/Pu-239	9.13E+01	5	5.48E+02	RPDT (LMITCO 1995a)
Cm-242/Pu-239	6.09E+00	5	3.65E+01	RPDT (LMITCO 1995a)
Cm-244/Pu-239	2.83E+00	5	1.70E+01	RPDT (LMITCO 1995a)

RPDT = Recent and Projected Data Task

TRA = Test Reactor Area

a. The relative standard deviations 0, 1, and 5 were obtained from the RPDT (LMITCO 1995a, Table 5-3 pp. 5 to 11).

b. Cotton, G. B., Interdepartmental Communication, September 2, 1999, "Estimate of Select Radionuclides Curie Content Generated by TRA Reactors and Disposed at the RWMC in Resins," GBC-01-00, Idaho National Engineering and Environmental Laboratory, Lockheed Martin Idaho Technologies Company, Idaho Falls, Idaho.

c. Schnitzler, G. B., Interdepartmental Correspondence to E. B. Nieschmidt, February 10, 1994, "Radioisotopes in ATR Fuel Elements," BGS-2-94, Idaho National Engineering and Environmental Laboratory, EG&G Idaho, Idaho Falls, Idaho.

2.3.2.6.2 Noncompactible Waste—The noncompactible waste stream consists of wood, metals, and glass, and is by far the largest waste stream by volume. However, this waste stream has a relatively small amount of activity associated with it. Scaling factors exist for this type of waste, specific to the TRA in the RPDT, and they were used to update the radionuclide content of this waste stream even though the overall contribution is relatively small. The scaling factors used to update the noncompactible waste stream are listed in Table 2-10.

Table 2-10. Noncompactible scaling factors for the Test Reactor Area.

Nuclide Scaling Ratio	Scaling Factor	Relative Standard Deviation ^a	Upper-Bound Scaling Factor	Scaling Factor Reference
H-3/Co-60	1.22E-01	5	7.32E-01	Best and Miller (1987)
C-14/Co-60	1.64E-03	5	9.84E-03	Best and Miller (1987)
Fe-55/Co-60	2.84E+00	1	5.68E+00	Best and Miller (1987)
Ni-59/Co-60	8.51E-04	5	5.11E-03	Evans et al. (1984)
Ni-63/Co-60	4.78E-01	1	9.56E-01	Best and Miller (1987)
C-60/Co-60	1.00E+00	1	2.00E+00	Best and Miller (1987)
Sr-90/Co-60	1.37E-03	5	8.22E-03	Best and Miller (1987)
Nb-94/Co-60	—	5	0.00E+00	Not available
Tc-99/Cs-137	9.00E-04	5	5.40E-03	Best and Miller (1987)
I-129/Cs-137	2.20E-07	5	1.32E-06	Not available
Cs-137/Cs-137	1.00E+00	1	2.00E+00	Best and Miller (1987)
Cs-137/Co-60	2.99E-01	1	5.98E-01	Best and Miller (1987)
Ce-144/Cs-137	2.35E-02	5	1.41E-01	Best and Miller (1987)
Eu-154/Cs-137	1.45E-05	5	8.70E-05	Evans et al. (1984)
Eu-155/Cs-137	4.70E-02	5	2.82E-01	Croff (1980) and Schnitzler (1994) ^b
U-234/Pu-239	3.90E-02	0	3.90E-02	Croff (1980) and Schnitzler (1994) ^b
U-235/Pu-239	8.33E-04	0	8.33E-04	Croff (1980) and Schnitzler (1994) ^b
U-236/Pu-239	1.48E-02	0	1.48E-02	Croff (1980) and Schnitzler (1994) ^b
Np-237/Pu-239	2.22E-02	5	1.33E-01	Croff (1980) and Schnitzler (1994) ^b
Pu-238/Pu-239	1.00E+00	5	6.00E+00	Best and Miller (1987)
Pu-239/Pu-239	1.00E+00	5	6.00E+00	Best and Miller (1987)
Pu-239/Co-60	8.06E-05	5	4.84E-04	Best and Miller (1987)
Pu-240/Pu-239	1.04E-01	5	6.24E-01	Croff (1980) and Schnitzler (1994) ^b
Pu-241/Pu-239	1.09E+02	5	6.54E+02	Best and Miller (1987)
Am-241/Pu-239	5.00E-01	5	3.00E+00	Best and Miller (1987)
Cm-242/Pu-239	5.00E-01	5	3.00E+00	Best and Miller (1987)
Cm-244/Pu-239	4.63E-01	5	2.78E+00	Best and Miller (1987)

a. The values shown were obtained from the Recent and Projected Data Task (LMITCO 1995a, Table 5-2, pp. 5 to 11).

b. Schnitzler, G. B., Interdepartmental Correspondence to E. B. Nieschmidt, February 10, 1994, "Radioisotopes in ATR Fuel Elements," BGS-2-94, Idaho National Engineering and Environmental Laboratory, EG&G Idaho, Idaho Falls, Idaho.

2.3.2.6.3 Core Structural Parts—The RWMC received seven shipments of ATR core structural parts, one in 1994 and six in 1995. Information on the physical contents and radionuclide inventory is contained in the following references, all of which are internal correspondence:

- “Advanced Test Reactor (ATR) Canal Trash Characterization for Trash Liner No. 51”ⁱ
- “Canal Trash Can No. 61”^j
- “Canal Trash Liner No. 75”^k
- “Rev. 1, Canal Trash Liner No. 74”^l
- “Canal Trash Liner No. 68”^m
- “Canal Trash Liner No. 62”ⁿ
- “Canal Trash Liner No. 65.”^o

Because scaling factors do not exist for this waste stream, a reassessment was made using the above references to estimate activities for the long-lived radionuclides (i.e., C-14, Ni-59, Ni-63, Co-60, Nb-94, Tc-90, and Sr-90) that were not included in the original assessments. The reassessment used data from Logan (see footnote g, p. 25). The estimated activity values for the above radionuclides were added to the database.

2.3.2.6.4 Hot Cell Waste—The hot cell waste, which consisted of one shipment in 1997, four shipments in 1998, and one in 1999, has unusual radioactive content. The 1997 shipment comprised high-efficiency particulate air (HEPA) filters from the hot cells, and Co-60 was the prevalent radionuclide. The radioactivity in the 1998 shipments was almost entirely from Eu-152, Eu-154, and Eu-155, and activity in the 1999 shipment was mostly from Ir-192, with significant contributions from Eu-152, Eu-154, Eu-155, and Co-60. The total activity of the hot cell waste stream is relatively small compared to the total from 1994 to 1999. Because the radionuclide content of the hot cell waste streams would depend entirely on the specific activities and processes carried out in the hot cell, and each activity

i. Brower, J. O., Interdepartmental Correspondence to L. J. Toomer, January 12, 1994, “Advanced Test Reactor (ATR) Canal Trash Characterization for Trash Liner No. 51,” JOB-01-94, Idaho National Engineering and Environmental Laboratory, EG&G Idaho Inc., Idaho Falls, Idaho.

j. Brower, J. O., Interdepartmental Communication to S. W. Bradley, March 7, 1995, “Canal Trash Can No. 61,” JOB-06-95, Idaho National Engineering and Environmental Laboratory, Lockheed Martin Idaho Technologies Company, Idaho Falls, Idaho.

k. Brower, J. O., Interdepartmental Communication to S. W. Bradley, April 12, 1995, “Canal Trash Liner No. 75,” JOB-09-95, Idaho National Engineering and Environmental Laboratory, Lockheed Martin Idaho Technologies Company, Idaho Falls, Idaho.

l. Brower, J. O., Interdepartmental Communication to S. W. Bradley, May 25, 1995, “Rev. 1, Canal Trash Liner No. 74,” JOB-18-95, Idaho National Engineering and Environmental Laboratory, Lockheed Martin Idaho Technologies Company, Idaho Falls, Idaho.

m. Brower, J. O., Interdepartmental Communication to S. W. Bradley, June 1, 1995, “Canal Trash Liner No. 68,” JOB-19-95, Idaho National Engineering and Environmental Laboratory, Lockheed Martin Idaho Technologies Company, Idaho Falls, Idaho.

n. Brower, J. O., Interdepartmental Communication to S. W. Bradley, July 18, 1995, “Canal Trash Liner No. 62,” JOB-21-95, Idaho National Engineering and Environmental Laboratory, Lockheed Martin Idaho Technologies Company, Idaho Falls, Idaho.

o. Brower, J. O., Interdepartmental Communication to S. W. Bradley, August 11, 1995, “Canal Trash Liner No. 65,” JOB-25-95, Idaho National Engineering and Environmental Laboratory, Lockheed Martin Idaho Technologies Company, Idaho Falls, Idaho.

or process may produce entirely different radionuclide contents, general scaling factors for hot cell waste would be difficult to generate. Therefore, no attempt was made to update the radioactive content of this waste stream. Contributions from the hot cell waste comprise slightly more than 1% of the total TRA activity listed in the IWTS database from 1994 to 1999.

2.3.2.6.5 New Production Reactor Irradiated Target Material—The RWMC received eight shipments, described as “NPR Irradiated Target Material (TRISO^p Target)” in the IWTS database, in 1997. With the exception of one shipment listed as having virtually no activity, the NPR target shipments each had several hundred curies of activity attributable almost entirely to tritium. Tiny amounts of Co-60, in the less than millicurie range, also are listed. Applying scaling factors based on Co-60 content, in this case, would make an insignificant contribution to the overall activity. Therefore, no attempt was made to calculate revised estimates of radioactivity for this unique waste stream.

2.3.2.6.6 Other—Similarly, a single shipment of sludge, described as “TRA-645 Cold Well Sludge” in the IWTS database, and two shipments of Portland cemented waste all have millicurie (or smaller) amounts of activity, and no effort was made to modify the listed amounts of radionuclides with scaling factors.

2.3.3 Test Area North

2.3.3.1 Waste Generator. Test Area North is located at the north end of the INEEL about 43.5 km (27 mi) northeast of CFA. Test Area North was established in the 1950s by the U.S. Air Force and Atomic Energy Commission Aircraft Nuclear Propulsion Program to support nuclear-powered aircraft research. Upon termination of this research, TAN facilities were converted to support a variety of other DOE research projects. Unless it becomes necessary for the United States to resume former levels of defense-related activities, the future of TAN will consist of completing current programs at the Technical Support Facility (TSF) and Contained Test Facility areas, deactivating all facilities, and completing ER activities. The designation recognized under the FFA/CO and CERCLA for TAN is WAG 1. Waste was generated during remedial investigation and ER activities at WAG 1.

Also located in the TAN area, west of the TSF, was the Loss-of-Fluid Test (LOFT) facility, a scaled-down version of a nuclear-powered utility generation station and the mobile test assembly, which contained the LOFT reactor vessel and the primary coolant system. This facility was used to simulate, to a limited extent, the conditions that existed in the Three Mile Island (TMI) Unit II generating station in March 1979.

The General Electric Aircraft Nuclear Propulsion Program hangar building (TAN-629) and other buildings were occupied by the SMC project starting in 1985. The project produced armor plate for U.S. Army tanks. The waste specific to SMC is discussed in Section 2.3.4.

Today, many of the facilities at TAN, such as the Initial Engine Test Facility, are being demolished to support ER activities. Other facilities are being used to support a variety of ongoing DOE programs, including the Spent Fuel Program. The Water Reactor Research Test Facility area is scheduled for a major rehabilitation to support ongoing research and development activities. Various TAN facilities, including the TAN Hot Shop, have been involved in the receipt, transfer, preparation, and interim storage of the TMI Unit II reactor core.

2.3.3.2 Generation of the Waste. Inventories of waste shipped from TAN to the RWMC, in the years 1994 to 1999, consisted primarily of waste generated from operations performed in the

p. TRISO is the acronym for tri-isotopic.

decontamination shop and acid pits. This waste was ultimately correlated to materials associated with light water reactors.

Various types of waste from programmatic and ER activities include the following:

- TSF-11 (clarifier) concrete
- TSF-20 (acid neutralization pits)
- Test Area North Hot Shop noncompactible waste
- Test Area North Hot Shop debris
- Contaminated soil
- Solidified LLW water
- Nonfuel-bearing pressurized water reactor components
- TAN-726 D&D&D material
- Dried domestic sewage sludge
- Low-level wood and metal.

The TMI fuel, fuel debris, and other spent fuel currently stored in the TAN pool are projected to be moved into dry storage at INTEC over the next 6 to 7 years (DOE 1995a). Waste to be generated during this operation will be disposed of at the RWMC.

2.3.3.3 General Availability of Information. The main source of data pertaining to the TAN waste shipments to the RWMC from 1994 to 1999 was the IWTS database. Various reports and consultations with personnel also provided information.

2.3.3.4 Data-Collection Approach. The approach used to collect data for the period between 1994 and 1999 involved reviewing the reported waste information and radioanalytical data in the IWTS database. All of the TAN IWTS data and information were downloaded onto a spreadsheet. The data were sorted by shipment year, waste type, and generating facility to arrive at the total waste volumes and radioactivity from each facility. The data also were sorted by waste containers and waste type. Waste stream characteristics were gathered from the IWTS database information.

2.3.3.5 Descriptions of Waste Streams. The TAN waste from 1994 to 1999 was divided into 22 waste streams (see Table 2-11).

2.3.3.6 Scaling-Factor Analysis. This section describes the methodology and rationale used to determine the scaling factors necessary to ascertain the radioactive inventory of activation products, fission products, thorium, uranium, and TRU radionuclides that probably were included, but not reported, in radioactive shipments made from TAN to the RWMC from 1994 to 1999.

The methodologies used to determine the most appropriate scaling factors for TAN waste are summarized below.

- Scaling factors were calculated from the actual radioanalytical data reported in the IWTS database for waste shipped from TAN to the RWMC from 1994 to 1999.

Table 2-11. Test Area North waste streams sent to the Radioactive Waste Management Complex from 1994 to 1999.

Waste Stream Identification Number	Waste Stream Description
TAN-WRR-1	Metal: steel and carbon
TAN-WRR-2	WRRTF-05 injection well and sediment
TAN-TSF-1	Sludge (sanitary sludge from TSF)
TAN-TSF-2	Wood, metal, steel, carbon, and concrete structural components
TAN-TSF-3	Absorbed liquids, noncombustible, and cement
TAN-TSF-4	TSF-20 acid neutralization pits
TAN-TSF-5	Clarifier (TSF-11) waste, concrete, and CERCLA waste
TAN-607-1	TAN Hot Shop debris, wiring and wiring devices, metal, and steel
TAN-607-2	TAN Hot Shop noncompactible waste
TAN-DRC-1	Dry rod consolidation project mock fuel assemblies
TAN-DFN-1	TAN decontamination shop radiation-contaminated asbestos and a filter (HEPA)
TAN-CTS-1	Contaminated soil
TAN-PWR-1	Nonfuel-bearing components and a pressure-water reactor
TAN-PCS-1	TSF-36 petroleum-contaminated soil (CERCLA)
TAN-WTR-1	Solidified LLW water, absorbed liquids, noncombustible clay
TAN-HGR-1	Soil from the mercury retort project
TAN-GWT-1	Generated low-level nonhazardous radioactive waste and ion exchange resin
TAN-650-1	LOFT mobile test assembly shield tank and double railroad car dolly
TAN-TAN-1	LLW wood and metal from TAN to be direct-disposed
TAN-603-1	TAN-603 boiler equipment—LLW
TAN-726-1	TAN-726 D&D&D material
TAN-623-1	Dried domestic sewage sludge from TAN TSF

CERCLA = Comprehensive Environmental Response, Compensation, and Liability Act

D&D&D = deactivation, decontamination, and decommissioning

HEPA = high-efficiency particulate air

LLW = low-level waste

LOFT = loss-of-fluid test

TAN = Test Area North

TSF = Technical Support Facility

- Scaling factors were calculated from the previous (1984 to 1993) inventory of radiological contaminants shipped from TAN to the RWMC (LMITCO 1995a).
- A list was compiled of applicable scaling factors derived and used by other reputable sources (e.g., EPRI, American Society of Mechanical Engineers, and TRA). The scaling factors most appropriate to TAN waste shipments were compared, evaluated, and applied.

2.3.3.6.1 Methodology I—The scaling factors for Methodology I were calculated from the actual radioanalytical data reported in the IWTS database for waste shipped from TAN to the RWMC from 1994 to 1999. All radionuclides were scaled to Co-60, Cs-137, and Pu-239, depending on whether they were activation products (which were scaled to Co-60, fission products (which were scaled to Cs-137), or TRU, thorium, and uranium isotopes (which were scaled to Pu-239). Because of the distinctiveness of the waste streams and the reported radionuclides, two scenarios (described below), with their accompanying advantages and disadvantages were evaluated.

2.3.3.6.1.1 Scenario A—Most individual waste streams reported in the IWTS database did not have an adequate number of radionuclides reported to provide a good statistical basis with which to calculate all the needed scaling factors. However, one waste stream was fairly well characterized and provided sufficient data to perform a statistical study and compute various scaling factors. This waste stream (identified as TANFAM940504L) consisted of 12 waste containers, each containing 22 reported radionuclides. The pros and cons of using this particular data and waste stream to calculate scaling factors are listed below.

- **Pro**—(1) Using data from one waste stream provided scaling factors for radionuclides that would not be obtainable from other waste streams and (2) the evaluation of this waste stream is thorough and statistically based
- **Con**—(1) The waste stream primarily comprised fission products and, therefore, does not provide adequate data for the activation products or TRU isotopes, and (2) the studied waste stream represents only approximately 0.0001% (1.97E-4 Ci) of the total activity reported for TAN in the IWTS database from 1994 to 1999 (i.e., 194.4 Ci) and, therefore, may not adequately represent all the other TAN waste streams.

2.3.3.6.1.2 Scenario B—The radioanalytical data for all TAN waste streams that were shipped to the RWMC from 1994 through 1999 were managed and evaluated as a single data set, according to radionuclide, using simple statistics (i.e., means [averages] and the standard deviations from the mean) from the RPDT. The pros and cons of using all the data for the waste streams to calculate scaling factors are listed below.

- **Pro**—The data encompass all the waste and provide a reasonable and practical representation of most activation products, fission products, or TRU, thorium, or uranium isotopes
- **Con**—Because the data represent many different types of waste and activity levels, some scaling factors have large variances (i.e., standard deviation from the mean) and may not provide good statistical defensibility.

Because the radionuclide scaling factors for Scenario B are higher (in most cases) than scaling factors associated with other sources, they will end up producing a more conservative (higher) inventory estimate. Consequently, any imprecision associated with the scaling factors would have a negligible impact to the overall estimate of the total radionuclide inventory at the RWMC. The manner in which the TAN IWTS data were evaluated (i.e., a simple, statistical, and mathematical approach) is adequate, considering the relatively small quantities of activity shipped from TAN as compared to the entire RWMC inventory (from all other INEEL facilities combined).

Individual radionuclide scaling factors (i.e., ratios) for Scenario B were calculated as follows:

$$SF_n = \frac{\text{Radionuclide result (curies) for each individual waste stream}}{\text{Scaling radionuclide result (curies) associated with each individual waste stream.}} \quad (16)$$

Examples:

$$SF_{C-14/Co-60} = \frac{\text{C-14 result for "nonfuel-bearing components"}}{\text{Co-60 result for "nonfuel-bearing components."}} \quad (17)$$

$$SF_{C-14/Co-60} = 1.84E-04 \text{ Ci} / 2.41E+01 \text{ C} = 7.63E-06. \quad (18)$$

Average (mean) radionuclide scaling factors (ratios) for Scenario B were calculated as follows:

$$SF_{\text{mean}} = \frac{(SF_1 + SF_2 + SF_3 + SF_n)}{n} \quad (19)$$

Upper-bound scaling factors for Scenario B were calculated as shown in the RPDT (LMITCO 1995a, Section 5).

Other scenarios also were evaluated for this methodology. A TAN waste stream, identified as “nonfuel-bearing components,” that accounted for the majority of the total activity shipped to the RWMC from 1994 to 1999 was evaluated separately from all the other TAN waste streams. The scaling factors determined from this scenario were inappropriately low, resulting from the relatively high levels of Co-60 associated with the waste stream. Also, this waste stream did not satisfactorily represent the fission products and TRU radionuclides associated with TAN waste. Scaling factors determined from other waste streams (exclusive of the nonfuel-bearing components) also did not provide a representative or realistic cross section of all the TAN waste. Consequently, this scenario was not considered in the final application of scaling factors.

2.3.3.6.2 Methodology II—The scaling factors for Methodology II were calculated from the previous (1984 to 1999) inventory of radiological contaminants shipped from TAN to the RWMC. The radionuclide data used to calculate scaling factors were extracted from the RPDT. All radionuclides were scaled to Co-60, Cs-137, and Pu-239, depending on whether they were activation products (which were scaled to Co-60), fission products (which were scaled to Cs-137), or TRU, thorium, or uranium (which were scaled to Pu-239) isotopes.

The scaling factors were calculated as follows:

$$SF = \frac{\text{Best-estimate radionuclide inventory result (curies)}}{\text{Best-estimate scaling radionuclide inventory result (curies)}} \quad (20)$$

Example:

$$SF_{C-14/Co-60} = \frac{\text{Best-estimate C-14 result}}{\text{Best-estimate Co-60 result}} \quad (21)$$

$$SF_{C-14/Co-60} = \frac{1.2E-02 \text{ Ci} = 1.88E-04}{6.4E+01 \text{ Ci}} \quad (22)$$

Upper-bound scaling factors for Scenario B were calculated as shown in the RPDT (LMITCO 1995a, Section 5).

2.3.3.6.3 Methodology III—Scaling factors derived and used by other reputable sources (e.g., EPRI, American Society of Mechanical Engineers, and TRA were compiled in Methodology III applied to waste streams similar to TAN dry waste streams, and were evaluated for comparative purposes.

2.3.3.6.4 Best-Estimate Scaling Factors—The SF_{bes} were determined from the scaling factors that could be identified as the most applicable and defensible (i.e., most representative of the waste stream, contained a reasonable number of data points for statistical evaluation, and would produce the most conservative [maximum] inventory result). The TAN waste streams contained no free-flowing liquids or known resins; therefore, all scaling factors were based on what can be considered dry waste. No decay corrections were applied to determine any TAN scaling factors because no detailed history of waste origination and generation was available.

2.3.3.6.5 Upper-Bound Scaling Factors—In addition to determining the SF_{beS} , SF^{up} s also were determined. The TAN waste data for 1994 to 1999 in the IWTS provided adequate data to perform a simple statistical analysis and determine a mean, and standard deviation from the mean, for all reported waste streams and radionuclides. The standard deviation was used to estimate the SF^{up} . The standard deviation was computed initially at the one-sigma confidence level (one standard deviation), and was changed to a two-sigma confidence level to determine the upper-bound values. The best estimate of an SF^{up} for the previous CIDRA scaling factors is defined in the RPDT.

The final scaling factors for TAN waste are shown in Table 2-12.

2.3.4 Specific Manufacturing Capability

2.3.4.1 Waste Generator. The SMC project, located at TAN, began operations in 1983 (Zagula 1995). The two primary facilities at TAN that support SMC are TAN-679, the hangar building originally used in the 1960s for the Aircraft Nuclear Propulsion Project, and TAN-629, a fabrication and assembly facility. Other facilities at TAN that provide various support activities include (1) TAN-602A, administrative offices; (2) TAN-606, maintenance facility; (3) TAN-607A, a research and development program plus a quality control program; (4) TAN-628, TAN TSF, waste storage, and housing; and (5) TAN-681, a drum evaporator unit, which was used to dry drums of nonacidic sludge but is no longer operational (LMITCO 1995a).

2.3.4.2 Generation of the Waste. The major operation at SMC is metal fabrication, using depleted uranium to make tank armor for the Abrams M1A2 tank. The fabrication is a semi-automated process in which metal is punched, laser-cut and sheared to precise specifications, and assembled into the final product. Construction of the tank armor is divided into two operations: (1) rolling operations in TAN-679 and (2) fabrication and assembly in TAN-629. During the course of these activities, depleted uranium waste is created, which contaminates areas within the facilities. Contaminated construction debris also is created during demolition and construction activities.

2.3.4.3 General Availability of Information. The main source of data pertaining to SMC waste shipments to the RWMC is the IWTS database. Additional information has been obtained from various technical reports. The information from these reports was used to assess the accuracy of the IWTS database and to evaluate scaling factors used in updating the radionuclide inventory of waste shipments sent to the RWMC. Reassessment of the scaling factors also relied on analytical laboratory reports from INTEC on the isotopic curie content of depleted uranium, (see footnote r below) as well as information obtained originally from Fernald Environmental Management Project in Fernald, Ohio.^{q,r}

2.3.4.4 Data-Collection Approach. The general data collection approach for the period 1994 to 1999 involved reviewing the data in the IWTS database. Additional information to generate scaling factors came from Waste Generator Services at TAN and from INTEC. Waste Generator Services supplied the information contained in Table 2-13. This information was based on analytical results obtained originally from Fernald Environmental Management Project (see footnote r below).

q. Fernald Environmental Management Project is a U.S. Department of Energy site 18 mi northwest of Cincinnati.

r. Sheldon, D. E., Interoffice Facsimile to J. Grande, November 7, 2000, "Major Isotopic Constituents in SMC Depleted Uranium-Bearing Wastes (Cu) Conversion Factors," Idaho National Engineering and Environmental Laboratory, Bechtel BWXT LLC, Idaho, Idaho Falls, Idaho.

Table 2-12. Final scaling factors for Test Area North dry waste.^a

Radionuclide Scaling Ratio ^b	Scaling Factors ^c	Relative Standard Deviation ^d	Scaling Factor Reference ^e
C-14/Co-60	1.88E-04	5	RPDT
Co-58/Co-60	1.56E-01	5	RPDT
Co-60/Co-60	1.00E+00	1	RPDT
Cr-51/Co-60	7.31E-02	5	TAN IWTS
Fe-55/Co-60	1.39E+00	1	TAN IWTS
H-3/Co-60	5.49E-01	5	TAN IWTS
Mn-54/Co-60	2.57E-03	5	TAN IWTS
Nb-94/Co-60	5.36E-05	5	TAN IWTS
Ni-59/Co-60	1.51E-03	5	TAN IWTS
Ni-63/Co-60	1.27E+00	1	RPDT
Tc-99/Co-60	2.19E-03	5	TAN IWTS
Ag-110m/Cs-137	2.71E-02	5	TAN IWTS
Ce-144/Cs-137	3.60E-02	5	TAN IWTS
Cs-134/Cs-137	3.32E-03	5	TAN IWTS
Cs-137/Cs-137	1.00E+00	1	RPDT
Eu-152/Cs-137	1.96E-03	5	TAN IWTS
Eu-154/Cs-137	7.34E-03	5	TAN IWTS
Eu-155/Cs-137	2.36E-04	5	TAN IWTS
I-129/Cs-137	6.76E-06	5	TAN IWTS
Ru-106/Cs-137	5.79E-04	5	TAN IWTS
Sb-125/Cs-137	3.59E-02	5	TAN IWTS
Sr-90/Cs-137	4.12E+00	5	TAN IWTS
Am-241/Pu-239	4.44E+00	5	RPDT
Cm-242/Pu-239	5.04E-01	5	TAN IWTS
Cm-244/Pu-239	1.43E-01	5	TAN IWTS
Np-237/Pu-239	7.41E-06	5	RPDT
Pu-238/Pu-239	1.18E+00	5	TAN IWTS
Pu-239/Pu-239	1.00E+00	1	RPDT
Pu-240/Pu-239	1.38E+00	5	TAN IWTS
Pu-241/Pu-239	6.95E+02	5	TAN IWTS
Th-228/Pu-239	2.52E-01	0	TAN IWTS
Th-230/Pu-239	8.14E-02	0	TAN IWTS
Th-232/Pu-239	4.91E-01	0	TAN IWTS
U-234/Pu-239	6.36E+01	0	TAN IWTS
U-235/Pu-239	1.60E+00	0	TAN IWTS
U-236/Pu-239	1.08E-04	0	TAN IWTS
U-238/Pu-239	2.73E+01	0	TAN IWTS

IWTS = Integrated Waste Tracking System

RPDT = Recent and Projected Data Task

TAN = Test Area North

a. No correction for radioactive decay has been applied to the values in this table.

b. The radionuclides shown were scaled to Co-60, Cs-137, or Pu-239, depending on whether they were activation products (Co-60), fission products (Cs-137), or transuranic, thorium, or uranium (Pu-239) isotopes.

c. Best-estimate scaling factors are the scaling factors (selected from the comparison table) that produced the highest (most conservative) estimated inventory for each radionuclide. The SF_{be} for the 1984 to 1993 Contaminated Inventory Database for Risk Assessment data were computed as follows: $SF = \text{best estimate (curies) for C-14} / \text{best estimate (curies) for Co-60} = SF_{C-14} = 1.2E-02 \text{ Ci} / 6.4E+01 \text{ Ci} = 1.88E-04$.

d. The relative standard deviations for each scaling factor were obtained from the Recent and Projected Data Task (LMITCO 1995a, Table 5-2).

e. The reference for the scaling factors included the Recent and Projected Data Task (LMITCO 1995a, Table 3-6b) and the Integrated Waste Tracking System database (i.e., the ratios of activity data in the database for TAN from 1994 to 1999).

Table 2-13. Major isotopic constituents in depleted uranium.^a

Isotope ^c	Depleted Uranium Quantity ^b		
	In 1 kg (Ci)	In 1 Ci (uranium isotopes) (Ci)	Ratio with U-238
U-238	3.35E-04	8.54E-01	1.00E+00
U-232	5.86E-07	1.49E-03	1.74E-03
U-234	4.60E-05	1.17E-01	1.37E-01
U-235	4.32E-06	1.10E-02	1.29E-02
U-236	6.33E-06	1.61E-02	1.89E-02
Tc-99 ^e	7.79E-08	1.99E-04	2.33E-04
Th-234	3.35E-04	8.54E-01	1.00E+00
Pa-234m	3.35E-04	8.54E-01	1.00E+00
Th-231	4.32E-06	1.10E-02	1.29E-02

a. This information in this table was derived from Waste Generator Services at Test Area North.

b. The measured isotopic content was provided from Fernald Environmental Management Project:

Sheldon, D. E., Interoffice Facsimile to J. Grande, November 7, 2000, "Major Isotopic Constituents in SMC Depleted Uranium-Bearing Wastes (Cu) Conversion Factors," Idaho National Engineering and Environmental Laboratory, Bechtel BWXT LLC, Idaho, Idaho Falls, Idaho.

The measured isotopic content and is based on analysis of samples completed in 1996 and 1998. The data represent the average of two analyses on each of three derbies (base metal).

c. The isotopes Th-234 and Pa-234 exist in equilibrium with U-238. The isotope Th-231 exists in equilibrium with U-235.

d. An analysis of Tc-99 content was conducted by the Idaho Nuclear Technology and Engineering Center in 1997. Four samples were obtained and analyzed with a Tc-99 content of 64.7 pCi/g, 33.8 pCi/g, 59.2 pCi/g, and 0 pCi/g for an average of 39.4 pCi/g. This compares with Fernald average results of 77 pCi/g. The table values are based on the Fernald results (see footnote b above).

As recently as late 1999, the Analytical Laboratory Department at INTEC also did some analytical tests on the depleted uranium used at SMC (Barg 2000). Analysis was done on 63 samples of depleted uranium and included estimated measurement uncertainties as well as the curie content of the radionuclides listed in Table 2-14. This analysis measured some of the radionuclides included in the Fernald data as well as four TRU radionuclides.

2.3.4.5 Descriptions of Waste Streams. The waste streams from SMC operations could be divided generally into the four categories listed in Table 2-15. In all cases, the waste is contaminated with depleted uranium and no other source of contamination is present. The bulk of waste shipments in 1994 and 1995 was evaporator sludge and unsolidified slag with some low-level contaminated waste. No waste shipments from SMC were made to the RWMC in 1996 and 1997. Except for one shipment of sandblast grit in 1998, all the 1998 and 1999 shipments consisted of low-level contaminated waste. Examination of the content codes in the IWTS database revealed that the waste stream comprised a variety of materials including metals, glass, nonhalogenated plastics, wood, soil, gravel, concrete rubble, brick, and similar trash. In mid-1999, content codes were no longer used in the IWTS database and the waste stream description was listed simply as "Depleted Uranium Contaminated Material."

Table 2-14. Isotopic activity of constituents in depleted uranium as measured by the Idaho Nuclear Technology and Engineering Center.

Isotope	In 1 kg (Ci)	Uncertainty (Ci)	Ratio with U-238	Ratio Uncertainty
Am-241	3.16E-09	2.79E-09	1.19E-05	1.05E-05
Pu-238	2.77E-10	1.08E-10	1.05E-06	4.08E-07
Pu-239/240	4.54E-10	1.25E-10	1.71E-06	4.72E-07
Np-237	1.82E-09	8.20E-10	6.87E-06	3.09E-06
Tc-99	1.53E-07	3.98E-08	5.77E-04	1.50E-04
U-234	5.16E-05	1.51E-05	1.95E-01	5.70E-02
U-235	3.32E-06	1.73E-07	1.25E-02	6.53E-04
U-236	1.56E-06	1.40E-07	5.89E-03	5.28E-04
U-238	2.65E-04	1.38E-08	1.00E+00	7.36E-05

Table 2-15. Specific Manufacturing Capabilities waste streams sent to the Radioactive Waste Management Complex from 1994 through 1999.

Waste Stream Number	Description
SMC-628-1	Nonacidic evaporator sludge
SMC-628-2	Unsolidified slag
SMC-990-1	Material contaminated with depleted uranium (i.e., metals, glass, and gravel)
SMC-629-2	Sandblast grit

2.3.4.6 Scaling-Factor Analysis. The information in Tables 2-13 and 2-14 has been combined to generate the scaling factors for depleted uranium in Table 2-16. The highest value from either of the tables was used to generate duplicate scaling factors. The U-233/U-238 scaling factor was calculated from U-233 data contained in the IWTS database for the years 1998 and 1999.

Where possible, measurement uncertainties have been used to calculate the RSD for the scaling factor. The RSD is the standard deviation divided by the mean. The assumed RSD has been set equal to 1.00 in cases where uncertainty data are not available. The uncertainty in the case of U-238 is four orders of magnitude smaller than the mean and has, therefore, been set equal to zero. The SF^{up}s have been calculated using a two-standard-deviation approach. The upper-bound scaling-factor calculation equals the scaling factor times one, plus twice the RSD.

Table 2-16. Final scaling factors for Specific Manufacturing Capabilities depleted uranium waste.

Nuclide Scaling Ratio	Scaling Factor	Relative Standard Deviation	Upper-Bound Scaling Factor	Scaling-Factor Reference
Am-241/U-238	1.19E-05	0.88	3.28E-05	Barg (2000) and Appendix A
Np-239/U-238	6.87E-06	0.45	1.31E-05	Barg (2000) and Appendix A
Pa-234m/U-238	1.00E+00	1.00	3.00E+00	Fernald (Sheldon 2000, Table 13) ^a
Pu-238/U-238	1.05E-06	0.39	1.87E-06	Barg (2000) and Appendix A
Pu-239/U-238	1.71E-06	0.28	2.67E-06	Barg (2000) and Appendix A
Tc-99/U-238	5.77E-04	0.26	8.77E-04	Barg (2000) and Appendix A
Th-231/U-238	1.29E-02	1.00	3.87E-02	Fernald (Sheldon 2000, Table 13) ^a
Th-234/U-238	1.00E+00	1.00	3.00E+00	Fernald (Sheldon 2000, Table 13) ^a
U-232/U-238	1.75E-03	1.00	5.25E-03	Fernald (Sheldon 2000, Table 13) ^a
U-233/U-238	1.30E-01	1.00	3.90E-01	IWTS ^b
U-234/U-238	1.95E-01	0.29	3.08E-01	Barg (2000) and Appendix A
U-235/U-238	1.29E-02	0.05	1.42E-02	Fernald (Sheldon 2000, Table 13) ^a
U-236/U-238	1.89E-02	0.09	2.23E-02	Fernald (Sheldon 2000, Table 13) ^a
U-238/U-238	1.00E+00	0.00	1.00E+00	Barg (2000) and Appendix A

IWTS = Integrated Waste Tracking System

a. Sheldon, D. E., Interoffice Facsimile to J. Grande, November 7, 2000, "Major Isotopic Constituents in SMC Depleted Uranium-Bearing Wastes (Cu) Conversion Factors," Idaho National Engineering and Environmental Laboratory, Bechtel BWXT LLC, Idaho, Idaho Falls, Idaho.

b. The values shown were calculated directly from the inventories for the two isotopes reported in the Integrated Waste Tracking System database for the years 1998 and 1999.

2.3.5 Idaho Nuclear Technology and Engineering Center

2.3.5.1 Waste Generator. Formerly known as the Idaho Chemical Processing Plant, INTEC is located 4.8 km (3 mi) north of the CFA. The plant is situated on about 210 acres (85 ha) that lie within the plant's perimeter fence. An additional 55 acres (22 ha) of the plant area lie outside the fence. The INTEC was chartered in 1953 to reprocess spent nuclear fuel to recover and recycle fissile uranium. The reprocessing involved dissolving the spent fuel in nitric and hydrofluoric acids and a solvent extraction system that used tributyl phosphate, hexone, and nitric acid to recover the uranium. The designation recognized under the FFA/CO and CERCLA for INTEC is WAG 3 (see Figure 1-1). Waste was generated during the CERCLA-driven remedial investigation activities at INTEC.

The current mission of INTEC is to receive and store spent nuclear fuels and radioactive waste, treat and convert waste, and develop new technologies for waste and waste management for DOE. This mission also included nuclear fuel reprocessing. However, in April 1992, reprocessing work was phased out. Facilities once dedicated to reprocessing work are being converted to a safe and stable shutdown condition while awaiting reuse or D&D&D.

2.3.5.2 Generation of the Waste. Liquid waste from the reprocessing activities was stored in the Tank Farm at INTEC. From 1963 to 2000, fluidized-bed waste calciners were used to convert the high-level liquid waste into dry, granular solids that are stored in underground stainless steel bins, called calcined solids storage facilities. The granulated solids will remain in the bins until a process can be developed to convert the solid waste into a nonleachable form, such as a glass, that can be shipped to a permanent radioactive-waste repository.

When DOE discontinued the mission of reprocessing spent nuclear fuel in 1992, the emphasis of the INTEC mission became storage and management of the spent nuclear fuel and calcination of the high-level liquid waste. The 1995 Settlement Agreement (DOE 1995b) between the State of Idaho, the U.S. Navy, and DOE requires (1) discontinuing use of all liquid waste tanks contained in pillar and panel vaults by 2009, (2) calcinating all sodium-bearing waste stored in the Tank Farm by 2012, and (3) treating all high-activity waste stored at INTEC for shipment out of the State of Idaho by 2035.

2.3.5.3 General Availability of Information. The IWTS database is the main source of data pertaining to INTEC waste shipments to the RWMC. Additional information has been obtained from various technical reports. The information from these reports was used to assess the accuracy of the IWTS database and evaluate scaling factors to be used in updating the radionuclide inventory of waste shipments sent to the RWMC.

2.3.5.4 Data-Collection Approach. The general data-collection approach used for the period from 1994 to 1999 involved reviewing the data in the IWTS databases. Additional information about scaling factors and upper-bound estimates was derived from the RPDT.

2.3.5.5 Descriptions of Waste Streams. The waste streams from INTEC operations are primarily the result of handling, processing, and storing spent nuclear fuel. High-level waste emanating from fuel-processing activities is not sent to the RWMC. The LLW is the result of incidental contamination from handling, processing, and storage activities and can generally be grouped into a single category, as shown in Table 2-17. All of the LLW is contact-handled, dry radioactive waste, and from descriptions in the IWTS database, is primarily the result of D&D&D activities.

Table 2-17. Waste streams sent to the Radioactive Waste Management Complex from the Idaho Nuclear Technology and Engineering Center from 1994 to 1999.

Waste Stream Number	Description
CPP-ALL-1P	Contaminated structural materials to include metal, concrete, bricks, soil, gravel, wood, and plastics. Concreted ash from Idaho Nuclear Technology and Engineering Center combustibles processed at the Waste Experimental Reduction Facility.

2.3.5.6 Scaling-Factor Analysis. The scaling factors used to update the IWTS database to best-estimate values for radionuclide inventory are shown in Table 2-18. These scaling factors are the same activity scaling factors used for the similar waste stream from TRA (i.e., dry radioactive waste) (see LMITCO 1995a, Table 5.) The justification for using these similar scaling factors is twofold. First, the source of the contamination is the same (i.e., fuel rods from TRA), or at least similar, if the fuel rods to be processed at INTEC came from another reactor. Second, the total amount of activity during the 5-year period being assessed is modest when compared to the overall activity inventory sent to the RWMC during the same period. The amount of effort required to research and calculate scaling factors, based on INTEC radiological measurements of low-level contaminated waste, would not justify any differences that might be found in the relatively small amount of radiological activity in this waste stream. The relative standard deviations used to calculate the upper bounds of radionuclide activity are discussed in Section 3.3.

Table 2-18. Dry active waste scaling factors for the Idaho Nuclear Technology and Engineering Center.

Nuclide Scaling Ratio	Scaling Factor	Relative Standard Deviation ^a	Upper-Bound Scaling Factor	Scaling Factor Reference
H-3/Co-60	1.22E-01	5	7.32E-01	Best and Miller (1987)
C-14/Co-60	1.64E-03	5	9.84E-03	Best and Miller (1987)
Fe-55/Co-60	2.84E+00	1	5.68E+00	Best and Miller (1987)
Ni-59/Co-60	8.51E-04	5	5.11E-03	Evans et al. (1984)
Ni-63/Co-60	4.78E-01	1	9.56E-01	Best and Miller (1987)
C-60/Co-60	1.00E+00	1	2.00E+00	Best and Miller (1987)
Sr-90/Co-60	1.37E-03	5	8.22E-03	Best and Miller (1987)
Nb-94/Co-60	Not available	5	0.00E+00	Not available
Tc-99/Cs-137	9.00E-04	5	5.40E-03	Best and Miller (1987)
I-129/Cs-137	2.20E-07	5	1.32E-06	Not available
Cs-137/Cs-137	1.00E+00	1	2.00E+00	Best and Miller (1987)
Cs-137/Co-60	2.99E-01	1	5.98E-01	Best and Miller (1987)
Ce-144/Cs-137	2.35E-02	5	1.41E-01	Best and Miller (1987)
Eu-154/Cs-137	1.45E-05	5	8.70E-05	Evans et al. (1984)
Eu-155/Cs-137	4.70E-02	5	2.82E-01	Croff (1980) and Schnitzler (1994) ^b
U-234/Pu-239	3.90E-02	0	3.90E-02	Croff (1980) and Schnitzler (1994) ^b
U-235/Pu-239	8.33E-04	0	8.33E-04	Croff (1980) and Schnitzler (1994) ^b
U-236/Pu-239	1.48E-02	0	1.48E-02	Croff (1980) and Schnitzler (1994) ^b
Np-237/Pu-239	2.22E-02	5	1.33E-01	Croff (1980) and Schnitzler (1994) ^b
Pu-238/Pu-239	1.00E+00	5	6.00E+00	Best and Miller (1987)
Pu-239/Pu-239	1.00E+00	5	6.00E+00	Best and Miller (1987)
Pu-239/Co-60	8.06E-05	5	4.84E-04	Best and Miller (1987)
Pu-240/Pu-239	1.04E-01	5	6.24E-01	Croff (1980) and Schnitzler (1994) ^b
Pu-241/Pu-239	1.09E+02	5	6.54E+02	Best and Miller (1987)
Am-241/Pu-239	5.00E-01	5	3.00E+00	Best and Miller (1987)
Cm-242/Pu-239	5.00E-01	5	3.00E+00	Best and Miller (1987)
Cm-244/Pu-239	4.63E-01	5	2.78E+00	Best and Miller (1987)

a. The values shown were obtained from the Recent and Projected Data Task (LMITCO 1995a, Table 5-2, p. 5-11).

b. Schnitzler, G. B., Interdepartmental Correspondence to E. B. Nieschmidt, February 10, 1994, "Radioisotopes in ATR Fuel Elements," BGS-2-94, Idaho National Engineering and Environmental Laboratory, EG&G Idaho, Idaho Falls, Idaho.

2.3.6 Naval Reactors Facility

2.3.6.1 Waste Generator. Located in the western part of the INEEL, NRF is about 23 km (14 mi) north-northeast of the RWMC. The NRF was established in 1950 when construction began on the prototype power plant for the first U.S. Navy nuclear-powered submarine, the USS Nautilus. This prototype, later named S1W, was developed to test the propulsion plant design and to train Navy personnel to operate reactors in preparation for duty on nuclear-powered submarines and ships in the U.S. naval fleet. Two additional naval reactor prototypes were subsequently built at NRF: A1W in 1957 and S5G in 1965. The basic mission of these other prototypes was the same as for the original prototype, to test propulsion plant designs and train Navy personnel. At this time, the three reactor prototypes have been shut down. The S1W plant was shut down in October 1989, the A1W was shut down in January 1994, and the S5G plant was shut down in May 1995 (LMITCO 1995b). The designation recognized under the FFA/CO and CERLA for NRF is WAG 8 (see Figure 1-1).

The Expended Core Facility (ECF), built at NRF in 1958, was designed to receive irradiated naval reactor fuel, perform examinations on the fuel elements, remove excess structural material from the fuel elements, and transfer the fuel elements to INTEC.

Currently, fuel storage is being constructed at NRF. Once the fuel storage construction is completed, fuel elements no longer will be transferred to INTEC and fuel currently stored at INTEC will be transferred back to NRF for storage. In addition, naval-fuel test specimens irradiated in other reactors, such as the ATR, have been received and examined at the ECF. The fuels are remotely handled underwater in the ECF water pits. Water serves as a transparent shielding medium in which a number of procedures can be carried out, including disassembling, cutting, sawing, milling, and visually examining various parts of the fuel elements. Some procedures, also carried out in hot cells at the ECF, are described in the HDT.

2.3.6.2 Generation of the Waste. Low-level waste was generated by the naval reactor prototypes as a result of decommissioning work. At the ECF, LLW is generated as a result of fuel examination work. The majority of the waste volume generated at NRF was noncompactible waste with very low levels of radioactivity from daily operations and the decommissioning of the ECF hot cells and the prototypes. The radioactivity associated with the waste is in highly corrosion-resistant metal structural materials removed during the naval fuel examinations. This material is loaded in metal containers or inserts that, in turn, fit into large shielded shipping casks. These casks are then taken to the RWMC where the containers or inserts are removed and buried (LMITCO 1995b).

2.3.6.3 General Availability of Information. The IWTS database is the main source of data pertaining to the waste from NRF through the time period from 1994 to 1999.

2.3.6.4 Data-Collection Approach. The data collection approach used included entering the NRF IWTS data information into a spreadsheet, and sorting the data by disposal year and waste type to arrive at the total volume of waste and total radioactivity, the waste-container types, and the waste types. Waste stream characteristics were gathered from IWTS, within the IWTS material profile.

2.3.6.5 Descriptions of Waste Streams. The NRF waste sent to the SDA from 1994 to 1999 is divided into six waste streams (see Table 2-19). Some of these waste streams were extensions of the waste streams identified in previous reports (i.e., RPDT and HDT).

Table 2-19. Waste streams generated at the Naval Reactors Facility from 1994 to 1999.

Waste Stream Number	Description of Waste
NRF-601-2	Contaminated soil, gravel, brick, and concrete rubble from the deactivation, decontamination, and decommissioning of the SIW evaporation pond.
NRF-618-6	Resin and resin containers removed from the Naval Reactors Facility prototypes.
NRF-618-7	Low-level compactible and noncompactible waste resulting from work at the prototypes, the ECF water pits, and the ECF hot cells.
NRF-618-8	Structural components removed from U.S. Navy nuclear fuel modules.
NRF-618-9	Concrete and concrete residue generated from decommissioning ECF hot cells and prototypes. Concrete and metal surfaces contain polychlorinated biphenyl in the form of dried paint.
NRF-618-AA	One-time waste stream of contaminated, radioactive oil solidified in Petroset.

ECF = Expanded Core Facility

2.3.6.6 Asbestos Calculation. As noted previously in Sections 1 and 2.2, disposal data were collected from two different databases: RWMIS and IWTS. The RWMIS recorded approximate disposal of asbestos based on percent volume of the waste. The IWTS recorded approximate disposal of asbestos, based on percent weight and percent volume of the waste, depending on the IWTS material profile. Therefore, the amount of asbestos was calculated in two different ways depending on whether amounts of asbestos were recorded as percent weight or percent volume.

2.3.6.6.1 Percent Volume—The percent volume of asbestos is calculated as follows:

$$\text{Best estimate for asbestos (g)} = [\text{percent volume of asbestos}] \times [\text{gross volume of container}] \times [\text{conversion of ft}^3 \text{ to m}^3] \times [\text{asbestos content}] \times [\text{conversion of lb to g}] \times [\text{density of asbestos}] . \quad (23)$$

The upper-bound value was obtained by multiplying 1.5 times the best estimate. This value was used to calculate the upper-bound value for asbestos in the HDT.

2.3.6.6.2 Percent Weight—The approximate percent weight of the asbestos was not recorded in the IWTS database, but a range was given for the amount of asbestos in the container. The mid-value of the range of the percent weight of the asbestos was used for the best estimate.

$$\text{Best estimate for asbestos (g)} = [\text{percent weight of asbestos}] \times [\text{gross weight of container, as recorded}] . \quad (24)$$

The upper or highest value range of the asbestos percent weight was used for the upper-bound value.

2.3.6.7 Scaling-Factor Analysis. The methodology is described in this section that used to calculate the radioactive inventory of activation products, fission products, or TRU, thorium, or uranium radioisotopes that probably were included, but not reported, in radioactive waste shipments made from NRF to the RWMC from 1994 to 1999. The information presented in this report updates the projected

data that were discussed in the RPDT. Much of the analysis presented below is similar to that discussed in Section 3.2.

The activity of unreported radionuclides was determined by multiplying isotope-dependent scaling factors with Co-60, Cs-137, or Pu-239 activities reported in the NRF waste shipments made to the RWMC. The NRF shipping data (which included the activities of Co-60, Cs-137, Pu-239, as well as other radionuclides) were obtained from the IWTS database.

The scaling factors estimated for the current analysis, and those determined from inventory data presented in the RPDT (LMITCO 1995a, Table 3-17b), are shown in Tables 2-20 through 2-22. The scaling factors listed in Table 2-20 apply to activation products that result from neutron activation of reactor hardware components (mainly fuel assembly upper-support structures). Those shown in Table 2-21 apply to fission products and actinides (including TRU, thorium, and uranium isotopes) resulting from waste items contaminated by fuel materials. The information shown in Table 2-22 applies to radioactive waste generated as a result of resin processing.

The purpose of scaling factors is to estimate the curie inventory of radionuclides that were not reported in the NRF waste shipments, but probably are present because of the nature of the material that has been irradiated or because similar radionuclides are known to be present. Finding C-14, Ni-59, Ni-63, Nb-94, and other similar activation products whenever Co-60 is reported is expected in the case of activated metals. Hence, a scaling factor computed for C-14 (e.g., SF [C-14]) can be used to estimate the inventory of C-14 based on the reported Co-60 inventory. This scaling factor can be written in mathematical terms as

$$\text{Inventory of C-14} = \text{C-14 [Ci]} = \text{SF [C-14]} \times \text{Co-60 [Ci]} . \quad (25)$$

For this report, all calculated inventory results are computed at the time of disposal; therefore, the scaling factor numbers also should be valid at the time of disposal.

Most scaling factors for radioactive isotopes are dependent on the amount of holdup or decay time of the waste prior to disposal at the RWMC. The decay time for all NRF waste shipments is estimated to be 5 years to account for the time required to remove the fuel from a naval reactor (e.g., a submarine, an aircraft carrier, or a land-based facility), package and ship this fuel to Idaho, process it at NRF, and ship the resulting LLW to the SDA. The holdup time for activated metals (including subassembly hardware) and other LLW represents an estimated average time because actual decay times are not known. The 5-year decay time assumed in this analysis represents a conservative value for determining all NRF scaling factors. Note that it is generally nonconservative to assume a zero decay time in scaling-factor calculations. The main reason for this is that scaling factors represent a ratio of two independent quantities (with the denominator being Co-60, in most cases). The Co-60 in the denominator usually decays faster than the radioisotope listed in the numerator. Therefore, the ratio of these two terms (i.e., the scaling factor) tends to increase with decay time.

The average amount of irradiation time for the naval reactor fuel and its associated hardware was assumed to be 10 years in some scaling-factor estimates. When no realistic information was available, engineering estimates of unknown reactor parameters had to be made to determine the SF_{be} for NRF waste. Thus, the information relative to the NRF fuel irradiation, decay, and material composition discussed in this report represents reasonable engineering estimates based on extrapolations of commercial PWR experience, knowledge of ATR, and other nonclassified information. Actual naval reactor fuel irradiation specifications are classified data, and not discussed in this report.

Table 2-20. Activation product scaling factors for Naval Reactors Facility hardware.^a

Primary Isotope	Isotope Half-Life (years)	CIDRA ^a Best-estimate Scaling Factor	CIDRA ^a Upper-bound Scaling Factor	Scaling Factors Based on NRF Reported Activity	ORIGEN2-Based Scaling Factors ^b	Updated Best-estimate Scaling Factor	Upper-Bound Scaling Factors	Isotope Ratio
C-14	5.73E+03	2.33E-04	5.71E-04	6.34E-04	1.41E-04	6.34E-04	1.552E-03	C-14/Co-60
Na-22	2.61E-00	N/A	N/A	0.00E+01	N/A	N/A	N/A	Na-22/Co-60
Cl-36	3.01E+05	N/A	N/A	5.64E-06	N/A	5.64E-06	N/A	Cl-36/Co-60
Cr-51	7.58E-02	4.05E-02	9.52E-02	1.79E-02	2.04E-19	1.79E-02	4.216E-02	Cr-51/Co-60
Mn-54	8.55E-01	9.05E-07	3.81E-06	4.01E-02	4.83E-03	4.01E-02	1.687E-01	Mn-54/Co-60
Fe-55	2.68E-00	5.95E-01	1.43E-00	9.03E-00	7.51E-01	9.03E-00	2.168E+01	Fe-55/Co-60
Fe-59	1.22E-01	N/A	N/A	1.63E-02	3.70E-14	1.63E-02	N/A	Fe-59/Co-60
Co-57	7.44E-01	N/A	N/A	1.22E-02	N/A	1.22E-02	N/A	Co-57/Co-60
Co-58	1.94E-01	8.33E-02	2.00E-01	2.51E-01	2.21E-07	2.51E-01	6.019E-01	Co-58/Co-60
Co-60	5.27E-00	1.00E-00	2.38E-00	1.00E-00	1.00E-00	1.00E-00	2.381E-00	Co-60/Co-60
Ni-59	7.60E+04	N/A	N/A	2.62E-02	2.58E-02	2.62E-02	N/A	Ni-59/Co-60
Ni-63	1.00E+02	1.12E-00	2.62E-00	2.78E-00	3.08E-00	3.08E-00	7.207E-00	Ni-63/Co-60
Nb-94	2.00E+04	N/A	N/A	4.38E-05	2.21E-03	2.21E-03	N/A	Nb-94/Co-60
Nb-95	9.58E-02	1.62E-01	3.81E-01	9.49E-02	4.43E-17	9.49E-02	2.233E-01	Nb-95/Co-60
Tc-99	2.13E+05	6.67E-08	1.52E-07	1.09E-04	1.86E-06	1.09E-04	2.482E-04	Tc-99/Co-60

CIDRA = Contaminant Inventory Database for Risk Assessment

N/A = not available—sufficient information was not available to determine a reasonable number

NRF = Naval Reactors Facility

ORIGEN2 = Oak Ridge Isotope GENeration and Depletion Code Version 2

a. Assumptions of 10 years of irradiation of the hardware components and 5 years of decay time prior to disposal at the Radioactive Waste Management Complex were incorporated into the ORIGEN2 results.

b. The values shown are ratios of activity data presented for the Naval Reactors Facility in the Recent and Projected Data Task (LIMITCO 1995a, Table 3-17b).

Table 2-21. A comparison of scaling factors for actinides and fission products for Naval Reactor Facility waste.

Primary Isotope	Isotope Half-Life (years)	CIDRA ^a Best-Estimate Scaling Factor	CIDRA ^a Upper-Bound Scaling Factor	Scaling Factors Based on NRF Reported Activity	ORIGEN2-Based Scaling Factor	Updated Best-estimate Scaling Factor	Updated Upper-bound Scaling Factor	Isotope Ratio
H-3	1.23E+01	5.60E+01	1.32E+02	1.40E+01	—	1.40E+01	3.310E+01	H-3/Cs-137
Sr-89	1.38E-01	N/A	N/A	—	N/A	N/A	N/A	Sr-89/Cs-137
Sr-90	2.90E+01	3.00E-00	7.20E-00	5.66E-01	—	5.66E-01	1.357E-00	Sr-90/Cs-137
Y-90	7.30E-03	3.00E-00	7.20E-00	5.50E-01	—	5.50E-01	1.321E-00	Y-90/Cs-137
Zr-95	1.75E-01	6.40E+04	1.54E+05	—	N/A	N/A	N/A	Zr-95/Cs-137
Tc-99	2.13E+05	N/A	N/A	—	See Tc-99 / Co-60	N/A	N/A	Tc-99/Cs-137
Ru-106	1.02E-00	N/A	N/A	3.39E-01	—	3.39E-01	N/A	Ru-106/Cs-137
Ag-110m	6.84E-01	N/A	N/A	—	N/A	N/A	N/A	Ag-110m/ Cs-137
Sn-113	3.15E-01	N/A	N/A	—	N/A	N/A	N/A	Sn-113/Cs-137
Sn-117m	3.72E-02	N/A	N/A	—	N/A	N/A	N/A	Sn-117m/Cs-137
Sb-124	1.65E-01	NA	N/A	—	N/A	N/A	N/A	Sb-124/Cs-137
Sb-125	2.76E-00	N/A	N/A	8.79E+02	—	8.79E+02	N/A	Sb-125/Cs-137
Te-132	8.92E-03	N/A	N/A	—	N/A	N/A	N/A	Te-132/Cs-137
I-129	1.60E+07	4.40E-05	1.60E-04	2.42E-05	—	2.42E-05	8.795E-05	I-129/Cs-137
I-131	2.20E-02	N/A	N/A	—	N/A	N/A	N/A	I-131/Cs-137
Cs-134	2.07E-00	N/A	N/A	1.76E-00	—	1.76E-00	N/A	Cs-134/Cs-137
Cs-137	3.02E+01	1.00E-00	2.20E-00	1.00E-00	—	1.00E-00	2.200E-00	Cs-137/Cs-137
Ba-140	3.49E-02	N/A	N/A	—	N/A	N/A	N/A	Ba-140/Cs-137
La-140	4.60E-03	N/A	N/A	—	N/A	N/A	N/A	La-140/Cs-137
Ce-144	7.80E-01	N/A	N/A	—	N/A	N/A	N/A	Ce-144/Cs-137
Eu-152	1.34E+01	N/A	N/A	5.57E-01	—	5.57E-01	N/A	Eu-152/Cs-137
Eu-154	8.50E-00	N/A	N/A	7.98E+01	—	7.98E+01	N/A	Eu-154/Cs-137
Eu-155	4.73E-00	N/A	N/A	5.15E+01	—	5.15E+01	N/A	Eu-155/Cs-137
Ta-182	3.13E-01	1.52E+05	3.60E+05	—	N/A	N/A	N/A	Ta-182/Cs-137
Pb-210	2.23E+01	N/A	N/A	No data	N/A	N/A	N/A	Pb-210/Cs-137
Ra-226	1.60E+03	N/A	N/A	No data	N/A	N/A	N/A	Ra-226/Cs-137
Ra-228	5.76E-00	N/A	N/A	No data	N/A	N/A	N/A	Ra-228/Cs-137
Ac-227	2.18E+01	N/A	N/A	No data	N/A	N/A	N/A	Ac-227/Pu-239

Table 2-21. (continued).

Primary Isotope	Isotope Half-Life (years)	CIDRA ^a Best-Estimate Scaling Factor	CIDRA ^a Upper-Bound Scaling Factor	Scaling Factors Based on NRF Reported Activity	ORIGEN2-Based Scaling Factor	Updated Best-estimate Scaling Factor	Updated Upper-bound Scaling Factor	Isotope Ratio
Th-228	1.91E+00	N/A	N/A	No data	N/A	N/A	N/A	Th-228/Pu-239
Th-229	7.30E+03	N/A	N/A	No data	N/A	N/A	N/A	Th-229/Pu-239
Th-230	7.54E+04	N/A	N/A	No data	N/A	N/A	N/A	Th-230/Pu-239
Th-232	1.40E+10	N/A	N/A	No data	N/A	N/A	N/A	Th-232/Pu-239
Pa-231	3.28E+04	N/A	N/A	No data	N/A	N/A	N/A	Pa-231/Pu-239
U-232	7.00E+01	N/A	N/A	No data	N/A	N/A	N/A	U-232/Pu-239
U-233	1.59E+05	N/A	N/A	No data	N/A	N/A	N/A	U-233/Pu-239
U-234	2.45E+05	N/A	N/A	1.00E-00	N/A	1.00E-00	N/A	U-234/U-234
U-235	7.04E+08	N/A	N/A	1.00E-00	N/A	1.00E-00	N/A	U-235/U-235
U-236	2.34E+07	N/A	N/A	No data	N/A	N/A	N/A	U-236/U-236
U-238	4.47E+09	N/A	N/A	1.00E-00	N/A	1.00E-00	N/A	U-238/U-238
Np-237	2.14E+06	N/A	N/A	No data	N/A	N/A	N/A	Np-237/Pu-239
Pu-238	8.77E+01	N/A	N/A	4.02E-00	N/A	4.02E-00	N/A	Pu-238/Pu-239
Pu-239	2.41E+04	N/A	N/A	1.00E-00	N/A	1.00E-00	N/A	Pu-239/Pu-239
Pu-240	6.56E+03	N/A	N/A	5.78E-01	N/A	5.78E-01	N/A	Pu-240/Pu-239
Pu-241	1.44E+01	N/A	N/A	4.65E+01	N/A	4.65E+01	N/A	Pu-241/Pu-239
Pu-242	3.76E+05	N/A	N/A	No data	N/A	N/A	N/A	Pu-242/Pu-239
Am-241	4.32E+02	N/A	N/A	6.51E-00	N/A	6.51E-00	N/A	Am-241/Pu-239
Am-243	7.37E+03	N/A	N/A	No data	N/A	N/A	N/A	Am-243/Pu-239
Cm-244	1.81E+01	N/A	N/A	2.77E-00	N/A	2.77E-00	N/A	Cm-244/Pu-239

CIDRA = Contaminant Inventory Database for Risk Assessment

N/A = not applicable

NRF = Naval Reactors Facility

ORIGEN2 = Oak Ridge Isotope GENeration and Depletion Code Version 2

a. The values shown are ratios of activity data presented for the Naval Reactors Facility in the Recent and Projected Data Task (LMITCO 1995a, Table 3-17b).

Table 2-22. Estimated scaling factors for resin materials (based on Abbott^a).

Primary Isotope	Isotope Half-Life (years)	NRF Resins Best-Estimate Scaling Factors	Upper-Bound Scaling Factors	Isotope Ratio
C-14	5.73E+03	0.0030	0.0420	C-14/Co-60
Na-22	2.61E-00	N/A	N/A	Na-22/Co-60
Cl-36	3.01E+05	N/A	N/A	Cl-36/Co-60
Cr-51	7.58E-02	N/A	N/A	Cr-51/Co-60
Mn-54	8.55E-01	N/A	N/A	Mn-54/Co-60
Fe-55	2.68E-00	N/A	N/A	Fe-55/Co-60
Fe-59	1.22E-01	N/A	N/A	Fe-59/Co-60
Co-57	7.44E-01	N/A	N/A	Co-57/Co-60
Co-58	1.94E-01	N/A	N/A	Co-58/Co-60
Co-60	5.27E-00	1.000	1.000	Co-60/Co-60
Ni-59	7.60E+04	0.004	0.011	Ni-59/Co-60
Ni-63	1.00E+02	0.250	1.420	Ni-63/Co-60
Nb-94	2.00E+04	N/A	N/A	Nb-94/Co-60
Nb-95	9.58E-02	N/A	N/A	Nb-95/Co-60
Tc-99	2.13E+05	1.30E-05	3.00E-05	Tc-99/Co-60
I-129	1.60E+07	5.20E-08	1.20E-07	I-129/Co-60

N/A = not applicable

NRF = Naval Reactors Facility

a. Abbott, M. L., Interdepartmental, to Mike Carbonneau, September 1997, "Revised Report on NRF Expended Resin Waste Activity Inventories," MLA-9-97, Idaho National Engineering Laboratory, Lockheed Martin Idaho Technologies Company, Idaho Falls, Idaho.

The assumed 5-year-decay time provides for a sufficient amount of time for many short-lived radionuclides to decay away, while having little effect on the calculated inventory of other long-lived radionuclides.

Time-dependent scaling factors SF(t)s can be computed from the initially estimated scaling factors (SF[0]s). Therefore, to determine the scaling factor for Ni-59 relative to Co-60 at some future time (t) where the initial scaling factor (SF[0]) is known or can be computed for some reference time (t = 0), the following applies:

$$SF[0] = [\text{initial activity of Ni-59 at } t = 0] / [\text{the initial activity of Co-60 at } t = 0]. \quad (26)$$

The time-dependent scaling factor SF(t) for Ni-59 is then computed as follows:

$$SF[t] = [\text{final activity of Ni-59 (t)}] / [\text{final Co-60 activity Co-60 (t)}]. \quad (27)$$

Equivalently,

$$SF[t] = \text{Ni-59}[t] / \text{Co-60}[t] = [\text{Ni-59}(0) \times \exp(-\lambda_1 t)] / [\text{Co-60}(0) \times \exp(-\lambda_2 t)] = SF(0) \times [\exp(-\lambda_1 t) / \exp(-\lambda_2 t)] \quad (28)$$

where

λ_1 = decay constant for Ni-59 (key isotope) = $\ln(2) / 7.6\text{E}+04 \text{ year} = 9.12\text{E}-06 \text{ per year}$

λ_2 = decay constant for Co-60 (reference isotope) = $\ln(2)/5.27 \text{ year} = 0.1315 \text{ per year}$.

Ni-59, with a decay time of $t = 5$ years, equals

$$\text{SF}(t) = \text{SF}(0) \times [\exp(-9.12\text{E}-06 \times 5) / \exp(-0.1315 \times 5)] = \text{SF}(0) \times 1.93 . \quad (29)$$

The general relationship is as follows:

$$\text{SF}(t) = \text{SF}(0) \times [\exp(-\lambda_1 t) / \exp(-\lambda_2 t)] . \quad (30)$$

The time-dependent scaling factor ($\text{SF}[t]$), in the case of Ni-59 (computed with a decay time of $t = 5$ years), is larger than the initial scaling factor ($\text{SF}[0]$) computed at $t = 0$ by a factor of 1.93. In this situation, scaling factors based on a zero-decay time ($t = 0$) will produce nonconservative radioactive inventories when calculated at the time of disposal, and for all radionuclides with half-lives longer than Co-60 (i.e., 5.272 years).

Several different techniques were used to determine the NRF scaling factors. Again, some of these techniques are similar to those already discussed for ANL-W (see Section 3.2). First, scaling factors (SF_{be} and SF^{up} values) were determined from previously reported inventory data shown in the RPDT. These scaling factors are shown in Columns 3 and 4 of Tables 2-20 and 2-21 for both metal hardware and fission products, respectively. The main purpose of the scaling-factor data, based on the RPDT information, is to provide a quality check on the other two techniques. The RPDT data also helped determine the SF^{up} used in this study.

A second set of scaling factors was determined from radioisotope inventory data already reported by NRF in the IWTS database. For example, the $6.34\text{E}-04$ values shown in Column 5 of Table 2-20 were determined through the following method. The inventories of both C-14 and Co-60 were reported for individual NRF waste shipments in the IWTS database. A curie ratio and a scaling factor were computed for each waste shipment that listed both terms, as follows: $\text{SF} = \text{C-14}(\text{curies})/\text{Co-60}(\text{curies})$. After all of these individual ratios were computed, an average of all the terms was then calculated. The average value is shown in Column 5 of Tables 2-20 and 2-21.

The third method involved the use of the ORIGEN2 computer code. The ORIGEN2 analysis was applied only to the activation of reactor hardware (i.e., subassembly upper structures). A corresponding ORIGEN2 analysis for the fuel was not possible. The details of the ORIGEN2 analysis are explained in Section 2.3.8.6.

2.3.6.8 ORIGEN2 Analysis. Four ORIGEN2 calculations were developed to model the upper structure region of a small PWR reactor (simulating a hypothetical naval reactor) to determine an independent set of scaling factors for NRF hardware components. Because the material composing the upper-structure region of naval reactors is classified, four possible metal alloys were considered. The metals that were investigated were Stainless Steel 304, Inconel X750, Inconel 718, and Inconel 600. The elemental impurity concentrations in these metals were determined by "Re-Assessment of the Elemental Compositions of Several Materials Used in ATR Hardware Composition," the "Assessment of Neutron-

s. Carboneau, M. L., Interdepartmental Communication to J. A. Logan, August 1, 1999, "Re-Assessment of the Elemental Compositions of Several Materials Used in ATR Hardware Components," MLC-03-99, Idaho National Engineering and Environmental Laboratory, Lockheed Martin Idaho Technologies Company, Idaho Falls, Idaho.

Activation Products in Low-Level Waste Discharged from Nuclear Reactors at the Test Reactor Area and Sent to the Radioactive Waste Management Complex for Disposal (see footnote e, p. 27), and from recent assay information on Inconel 718 (see the Shiva Technologies correspondence in Appendix C). A flux of $2.63\text{E}+13$ n/cm²/second was assumed for the upper structure region of the reactor and was applied for an irradiation period of 10 years (see the November 16, 2000, ORIGEN2 modeling results provided in Appendix C). Because the purpose of this analysis was to generate scaling factors (e.g., C-14/Co-60 and Ni-59/Co-60), the isotope ratio results were nearly independent of the assumed flux. The results of the ORIGEN2 calculations are shown in Figures 2-1 through 2-6. As can be seen in these figures, the scaling-factor ratios increase linearly with time for each metal and the maximum curve (at 10 years) was dependent on the assumed metal. The ORIGEN2 results do not show the final 5-year decay time imposed at the end of the 10-year irradiation period. The 5-year decay time was computed separately.

As an example, consider the ratio of C-14/Co-60 results shown in Figure 2-1. The peak scaling factor for C-14 is $7.33\text{E}-05$, which occurs at 10 years of irradiation for stainless steel 304. Therefore, the C-14 scaling factor at $t = 0$ is $SF(0) = 0.0000733$. The C-14-scaling factor relative to Co-60 at 5 years of decay is as follows:

$$SF(5 \text{ year}) = SF(0) \times [\exp(-\lambda_1 t) / \exp(-\lambda_2 t)] = 0.0000733 \times [0.999395 / 0.518204] = 1.41\text{E}-04 . \quad (31)$$

This value is then listed in Column 6 of Table 2-20 and represents the estimated scaling factor based on the ORIGEN2 calculations. The other numbers in Column 6 of Table 2-20 were computed in a similar fashion. Entries with "NA" simply mean that sufficient information was not available to determine a reasonable number. A comparison of the ORIGEN2 scaling factors (Column 6) and the RPDT data in Column 3 shows good agreement for C-14, Fe-55, and Ni-63 nuclides, and poor agreement for Cr-51, Co-58, and Nb-95. ORIGEN2 scaling factors are lower than the CIDRA numbers for Cr-51, Co-58, and Nb-95. The primary reason for the lower scaling factor is that all of these nuclides have relatively short half-lives. The assumed 5-year decay period for the ORIGEN2-based scaling factors allows all of these isotopes to decay to near 0, while the RPDT analysis does not account for this.

2.3.6.8.1 Best-Estimate Scaling Factors—The SF_{be} s (shown in Column 7 of Tables 2-20 and 2-21) were determined as the maximum of the two sets of data shown in Columns 5 and 6 of these tables. This selection is slightly conservative and an alternate selection might be justified using a different combination of the data shown in these tables. However, so many unknowns are associated with the NRF waste that this methodology, though slightly conservative, is expected to be reasonable. The SF_{be} for the NRF resin material was selected from information reported in Abbott.^t Most of the curie inventories of waste sent to the RWMC from NRF are associated with activated metals and not resin materials. Therefore, no additional efforts were expended to refine the analysis performed in 1997 (see footnote t below).

t. Abbott, M. L., Interdepartmental Communication, to Mike Carboneau, September 1997, "Revised Report on NRF Expended Resin Waste Activity Inventories," MLA-9-97, Idaho National Engineering Laboratory, Lockheed Martin Idaho Technologies Company, Idaho Falls, Idaho.

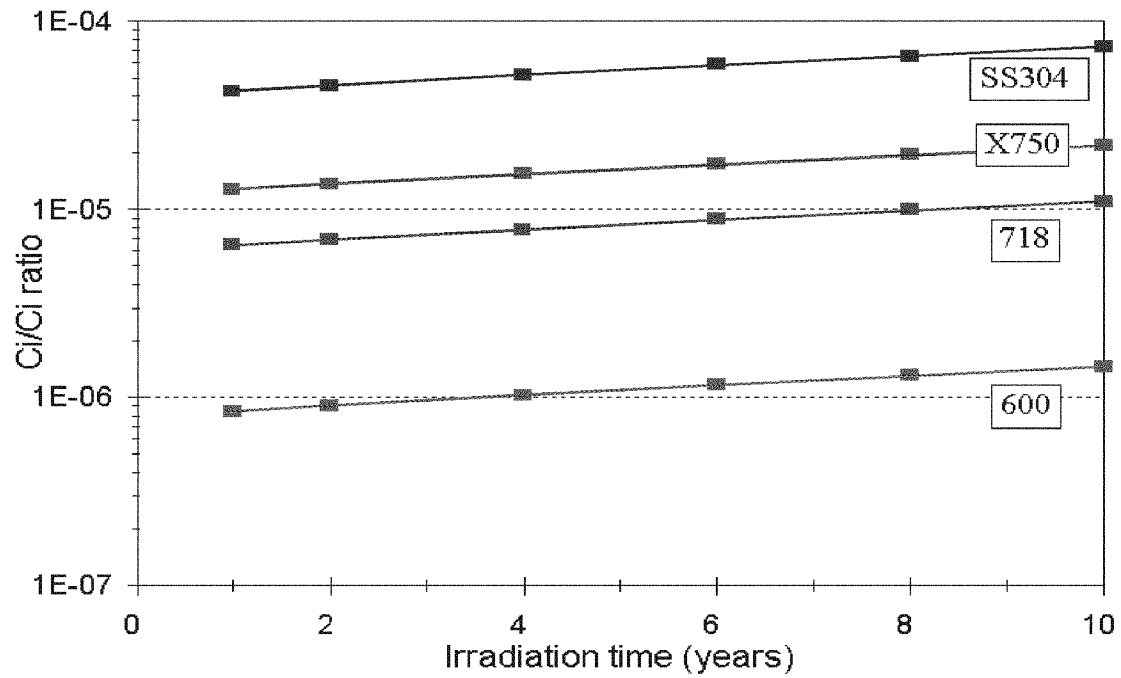


Figure 2-1. C-14/Co-60 (flux = 2.63E+13).

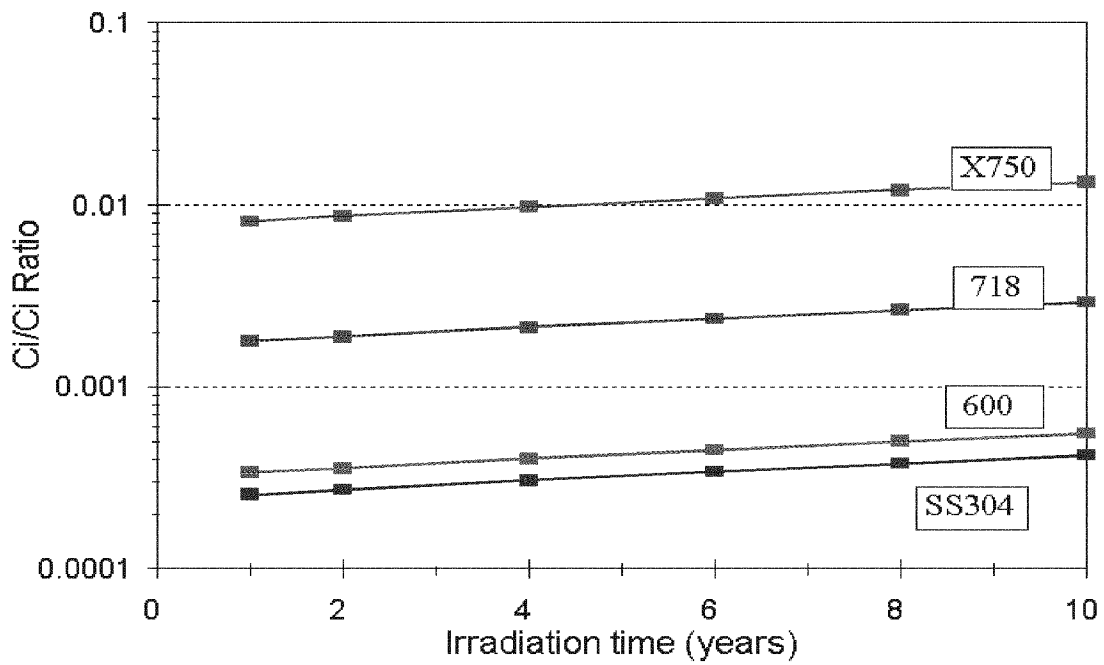


Figure 2-2. Ni-59/Co-60 (flux = 2.63E+13).

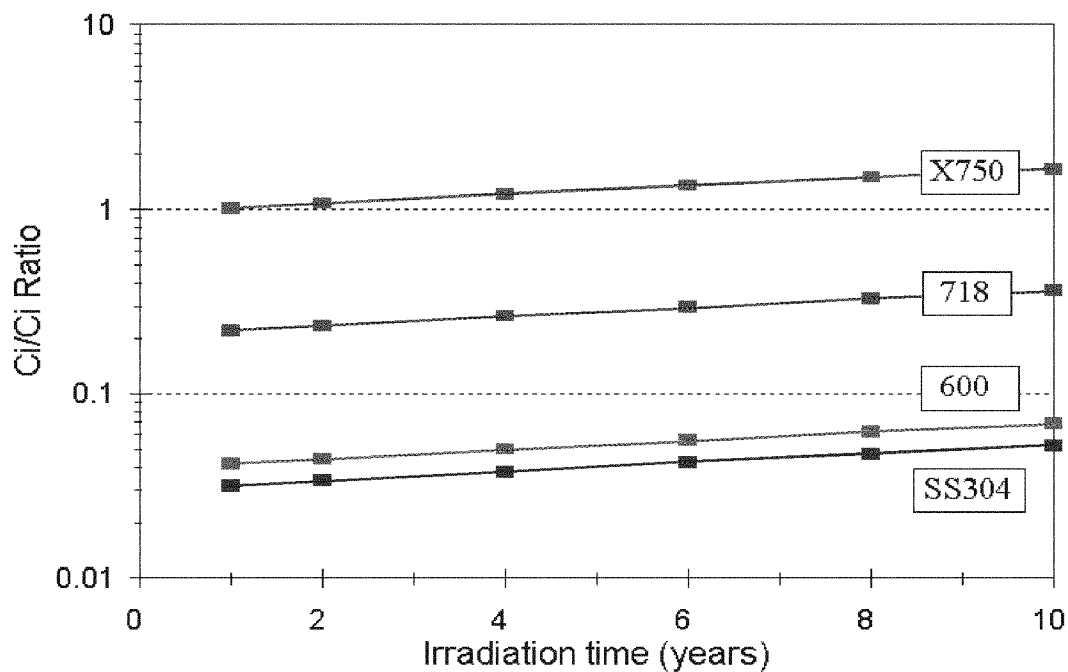


Figure 2-3. Ni-63/Co-60 (flux = 2.63E+13).

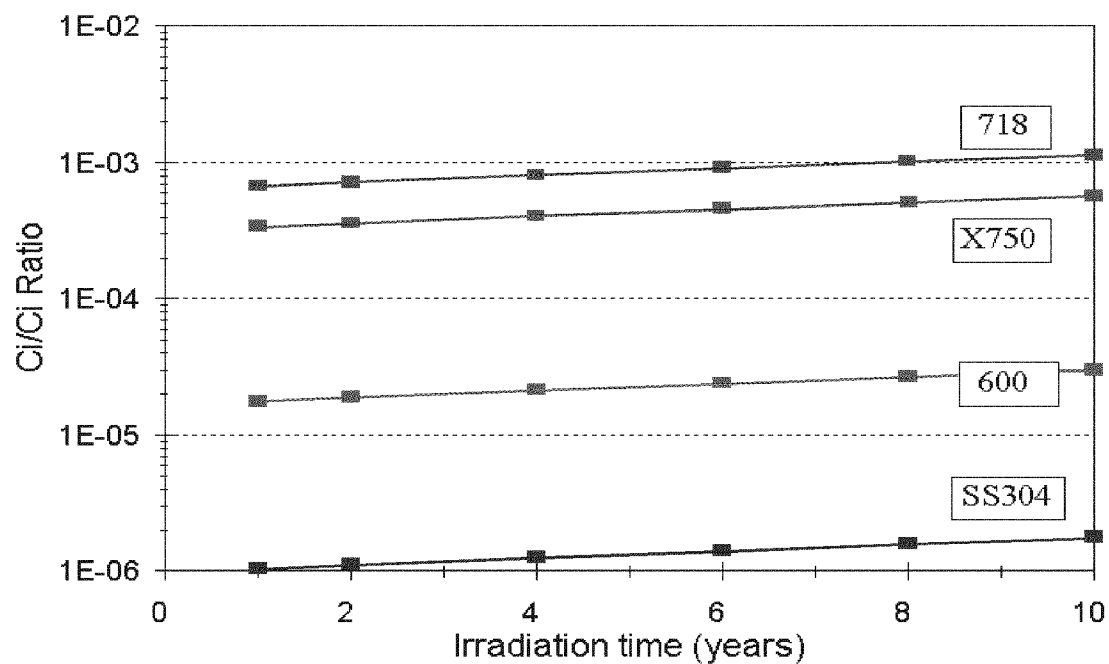


Figure 2-4. Nb-94/Co-60 (flux = 2.63E+13).

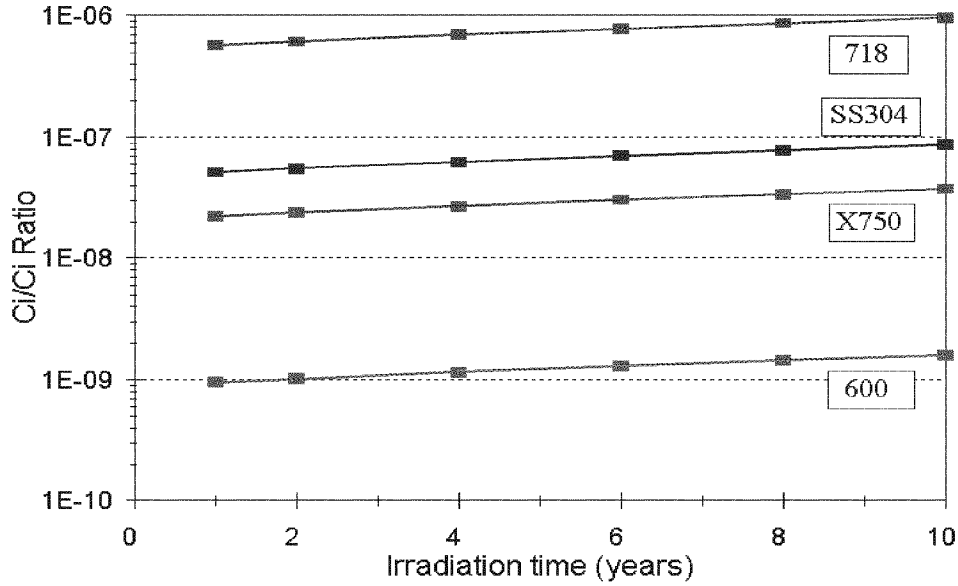


Figure 2-5. Tc-99/Co-60 (flux = 2.63E+13).

2.3.6.8.2 Upper-Bound Scaling Factors—In addition to calculating SF_{be} , the SF^{up} s were determined for activation products (i.e., metal waste) and waste contaminated with fission products. The purpose of the SF^{up} calculation was to determine a reasonable maximum inventory of several key isotopes that were not reported in the original shipping manifests, but are expected to be present in this waste.

The original intent of the SF^{up} was to determine inventory information by multiplying these scaling factors by the reported activity of a reference isotope. However, in actual practice, a variation of this technique was developed to simplify the mathematics associated with applying this information to the spreadsheet program. In actual practice, the upper-bound inventories were determined as follows:

$$[\text{Upper-bound inventory (curies)}] = [\text{best-estimate inventory (curies)}] \times [SF^{up}/SF_{be}] . \quad (32)$$

The best-estimate curie inventory was either a reported value or was calculated using the following relationships:

$$[\text{Best-estimate inventory (curies)}] = [\text{reference isotope activity (Co-60, Cs-137, or Pu-239)}] \times SF_{be} . \quad (33)$$

Note that the upper-bound inventory can be rewritten as follows:

$$[\text{Upper-bound inventory (curies)}] = [\text{reference isotope activity (Co-60, Cs-137, or Pu-239)}] \times SF^{up} = [\text{best-estimate inventory} / SF_{be}] \times SF^{up} = [\text{best-estimate inventory}] \times [SF^{up}/SF_{be}] . \quad (34)$$

Unlike the technique used to generate the SF_{be} , no simple method existed for determining the SF^{up} for every possible isotope or situation. In some cases, the information was inadequate to determine any SF^{up} , and in these situations, a result of “N/A” was listed in Table 2-20. In general, the basic technique used to determine the SF^{up} relied on the past work performed for the NRF facility, as documented in the RPDT (LMITCO 1995a, Table 3-17b). When RPDT data were available for a particular isotope (e.g., Columns 3 and 4 of Tables 2-20 and 2-21), then the SF^{up} for the current analysis was determined as simply

$$SF^{up} = SF_{be} \times (RPDT SF^{up}) / (RPDT SF_{be}) . \quad (35)$$

For example, the RPDT SF^{up} from Table 2-20, Column 4, for C-14 equals $5.71E-04$. The corresponding RPDT SF_{be} from Column 3 of Table 2-20 is $2.33E-04$. Therefore, the SF^{up} for the updated analysis is computed as $6.34E-4 \times (5.71E-4 / 2.33E-4) = 1.55E-3$, which is shown in Column 8 of Table 2-20. When the RPDT scaling factors were not known, an entry of “N/A” was reported. However, this was not acceptable for determining the inventories of those nuclides that had an SF_{be} . When no SF^{up} was assigned, then an “N/A” table value was equivalent to assuming an SF^{up} of zero in the spreadsheet analysis. Under these conditions, the spreadsheet analysis would produce an upper-bound inventory smaller than the best-estimate inventory. Making an assumption that the SF^{up} s were equal to the SF_{be} s was necessary to correct for this problem, and to produce upper-bound activities at least as big as the calculated best-estimate activities. An assumption made in the spreadsheet analysis was that $SF^{up} = SF_{be}$ whenever $SF^{up} = \text{“N/A”}$ in the table listings, in these special cases.

2.3.6.9 Estimated Inventories of Radioisotopes in Naval Reactors Facility Waste Shipments. After the SF_{be} and SF^{up} were determined, curie inventories of radioisotopes that were probably present, but not reported, in the NRF waste shipments to the RWMC from 1993 to 1999 could be calculated. When a radioisotope was listed in the IWTS database, then no scaling-factor analysis was necessary. The reported data were accepted as reported. However, when a key isotope was not reported in the waste shipment records, the activity of the isotope could be determined by multiplying the appropriate scaling factor (e.g., SF_{be} or SF^{up} obtained from Table 2-4) by its reference isotope activity (i.e., Co-60, Cs-137, or Pu-239).

The estimated inventories of radionuclides present in the NRF waste shipments (at the time of disposal) were determined as follows:

- When the particular radionuclide was listed in the database record, the inventory value of the radionuclide was accepted as was, with no change.
- When the radionuclide was not listed, but a reference isotope was reported (e.g., Co-60), then the best-estimate inventory of the radionuclide was computed using a scaling factor and the reported inventory for the reference isotope. All scaling factors and reported inventory data were determined at the time of disposal.
- The upper-bound inventories were determined based on the reported or calculated best-estimate inventories. Initially, a simple use of the SF^{up} (see Tables 2-20, 2-21, and 2-22) and the reported inventory of the reference isotope appeared appropriate rather than use of the best-estimate inventory. However, in some cases, a mathematical simplification required modification of the previous technique. Therefore, the previously calculated best-estimate inventory result was multiplied by a ratio of SF^{up} and SF_{be} to obtain the upper-bound inventory estimates, as shown below.

$$[\text{Upper-bound inventory for Ni-59}] = [\text{best-estimate inventory for Ni-59}] \times [SF^{up}(\text{Ni-59}) / SF_{be}(\text{Ni-59})] . \quad (36)$$

This formula is equivalent to the direct approach. However, the revised mathematical approach actually simplifies the logic in the spreadsheet formulas. For example,

$$[\text{Estimated upper-bound inventory for Ni-59}] = [\text{Co-60 inventory}] \times SF^{up}[\text{Ni-59}] = [(\text{Co-60 inventory}) \times SF_{be}(\text{Ni-59})] \times [SF^{up}(\text{Ni-59}) / SF_{be}(\text{Ni-59})] = [\text{best-estimate inventory for Ni-59}] \times [SF^{up}(\text{Ni-59}) / SF_{be}(\text{Ni-59})] . \quad (37)$$

The advantage of the last formula is that either a calculated best-estimate inventory or a reported best-estimate inventory can be used to compute the upper-bound estimate for Ni-59. Also, in the case when no SF^{up} is known (e.g., $SF^{up} = N/A$), the ratio of the two scaling factors (i.e., SF^{up} and SF_{be}) is set equal to one. This approach is equivalent to setting the SF^{up} equal to the SF_{be} for those cases for which no other information is known. This assumption allows for the calculated upper-bound inventories to be at least as big as the best-estimate values. The alternative would be to use no SF^{up} (i.e., setting $SF^{up} = 0$) for those cases with “N/A” entries. However, this is the worst-case situation. If no SF^{up} is assumed when an SF_{be} is listed, then the calculated best-estimate inventories can exceed the upper-bound estimates.

2.3.7 Other Waste Generators

Waste generators other than those described in Sections 3.2 through 3.7 were evaluated from 1994 to 1999 including the following six on-Site generators: ARA, CFA, D&D&D, PBF, RWMC, and WERF. Some of these generators produced more than one waste stream, but the total radioactivity was very small. This small amount of radioactivity represented less than 0.15% of the total radioactivity received at the RWMC for burial in the SDA from all waste generators between 1994 and 1999.

2.3.7.1 Auxiliary Reactor Area—The ARA is located in the south-central portion of the INEEL. The ARA consists of four separate operational areas designated as ARA-I, ARA-II, ARA-III, and ARA-IV. The ARA-I and ARA-II facilities were constructed in 1957. Activities at the two facilities consisted of hot cell operations, materials research, and laboratory operations including sample preparation and inspection. Numerous minor structures such as a guardhouse, well house, chlorination building, decontamination and laydown building, power-extrapolation building, electrical substation, and several storage tanks also were part of the site. The ARA is located within WAG 5, the designation recognized under the FFA/CO and CERCLA for the ARA and PBF (see Figure 1-1). The ARA-I and ARA-II facilities were shut down formally in 1988 and 1986, respectively. Decontamination and complete dismantlement were initiated in 1995 and are nearing completion.

Construction of the ARA-III facility was completed in 1959 to house the Army Gas Cooled Reactor Experiment research reactor. Following deactivation, the facility was modified to support other ongoing tests until late 1965, when the Army Reactor Program was phased out. Two buildings were constructed at ARA-III in 1969 to provide additional laboratory and office space to support other INEEL programs. The ARA-III facility was shut down in 1989. Decontamination and complete dismantlement were initiated in 1990 and completed in 1999.

The ARA-IV facility was built to accommodate the Mobile Low Power Reactor 1, an active project from 1957 to 1964. The Nuclear Effects Reactor was operated at ARA-IV from 1967 to 1970. The area was closed down until 1975, at which time it was used temporarily for some welding qualification work. Decontamination and dismantlement were performed in 1984 and 1985. Since 1985, the area has been used occasionally for testing explosives in powdered-metal manufacture experiments. A small control building, a bunker, the buried remains of two leach pits, and a sanitary waste system are all that remain.

According to projections in the INEEL (2001) Comprehensive Facility and Land Use Plan, the ARA will be encompassed by a future buffer to public roads and will not be reused for future INEEL operations.

2.3.7.2 Central Facilities Area—The CFA is the INEEL Site primary support area and is located in the south-central portion of the INEEL. Some of the facilities in use at CFA were built in the 1940s and 1950s, making the average age of buildings at CFA 28 years. Of these buildings, 27% are recommended for excess or demolition, and another 7% are in poor condition. Recent completion of some newer structures in the area has upgraded the infrastructure somewhat and has helped to make overall operations

more efficient. Other modifications are continually in the planning stages to help meet the changing needs of the INEEL. Eighty percent of the current activities at CFA consist of INEEL-wide programmatic support such as transportation, maintenance, capital construction, environmental and radiological monitoring, security, fire protection, warehouses, sanitary landfill, training, medical services, receiving and storage, calibration laboratories, and a cafeteria. A small amount of research and development work also is conducted at CFA. Small amounts of waste are produced from these various operations. The designation recognized under the FFA/CO and CERCLA for the CFA is WAG 4.

2.3.7.3 Deactivation, Decontamination, and Decommissioning Program—The INEEL D&D&D Program was established in late 1977 and is still active. The D&D&D Program envelops surplus facilities located at TAN, TRA, INTEC, CFA, PBF, ARA, and the reactor experimental areas located near the RWMC. Areas of the INEEL assigned to ANL-W and NRF are excluded from this plan. Of the original 45 contaminated facilities originally identified as surplus, 27 have been decommissioned to date. The D&D&D Program has demolished more than 100 buildings and structures during the last 4 years. Current planning includes some aspect of D&D&D for more than 200 facilities at the INEEL over the next 10 years. Areas subjected to D&D&D activities that contributed to the waste buried at the SDA from 1994 to 1999 include the ARA, the CFA old laundry facility, EBR-I, and one unidentified building.

2.3.7.4 Power Burst Facility—The PBF is located within WAG 5 in the south-central portion of the INEEL Site, about 8 km (5 mi) east of CFA. Part of the Waste Reduction Operations Complex (WROC), PBF was originally known as the Special Power Excursion Reactor Test (SPERT) area. Four SPERT reactors were built, beginning in the late 1950s, as part of an early investigation involving reactor transient behavior tests and safety studies of light-water-moderated, enriched-fuel reactor systems. Currently, all of the reactors have been removed and most of the facilities have undergone D&D&D. The PBF presently consists of the PBF reactor area (north of SPERT-I), the PBF control area, WROC Lead Storage Facility (at the SPERT-II site), WERF at the SPERT-III site, and the Mixed Waste Storage Facility (SPERT-IV). Waste from WERF is reported separately from waste from PBF.

The PBF portion of WROC consists of the reactor area and that portion of the control area not used for WROC support. Work is being conducted in the reactor area and control area to resolve issues involving waste remediation and RCRA. Currently, the reactor area is used to store nuclear fuel while the control area is used for office space to support fuel and waste storage activities. Once the fuel has been removed from the reactor area (the fuel was removed in FY 2001), the reactor area and associated structures and buildings in the control area will undergo decontamination and dismantlement.

2.3.7.5 Waste Reduction Operations Complex—The WROC is located in the south-central portion of the INEEL Site, about 8 km (5 mi) east of CFA. Currently, WROC has three main facilities dedicated to waste management activities including (1) WERF, (2) the Mixed Waste Storage Facility, and (3) the Waste Engineering Development Facility/WROC Lead Storage Facility. The original missions of WROC were to test the operational behavior of nuclear reactors and to study the safety of light water-moderated, enriched-fuel systems. The current mission of WROC is to reduce the volume of LLW and mixed waste so that less space is required for burial in the SDA. As a result, WROC, located within WAG 5, receives waste from all INEEL facilities for treatment as well as from some off-Site facilities. The WROC is equipped with a 0.5-million Btu/hour incinerator, a 200-ton (203,209-kg) compactor, and a 27.2-m³ (960-ft³) sizing shop where waste can be cut to size. The facility also contains an off-gas system, ash-solidification room, indoor mixed-waste storage, and a 130.3-m³ (4,600-ft³) outdoor radioactive waste storage area.

The incineration process handles combustible waste with volume reductions averaging about 200:1. The ash is disposed of in the SDA without further treatment if it passes the U.S. Environmental Protection Agency toxicity characteristic leaching procedure (40 CFR 261). If it does not pass this test, it

is stabilized with Portland cement before disposal. The waste-sizing process relies on manually operated torches and mechanical cutting devices including saws and shears. Both metallic and wood structures are sized (the average size reduction is 5:1). The compaction process uses a unit with a compaction force of 200 tons. The waste is compacted into metal containers for disposal in the SDA. The average size reduction for this process is 5:1. At present, only waste sizing and compaction are in operation at WERF. Waste incineration has been shut down at WROC, and restart is not expected based on construction of the Advanced Mixed Waste Treatment Facility.

2.3.7.6 Radioactive Waste Management Complex—Small amounts of waste are generated by operations at the RWMC itself. As shown in Figure 2-6, the RWMC, located 11.3 km (7 mi) southwest of the CFA at the INEEL, is divided, by function, into four separate areas:

- The administrative area, located in the northeast section of the facility, consists of buildings used for office space and other activities that support operations.
- The operations zone, located west of the administrative area, consists of buildings and storage sheds used for operations and maintenance activities that support the RWMC.
- The SDA, a 97-acre (39-ha) area located in the western section of the facility, is dedicated to permanent, shallow-land disposal of solid LLW. The SDA contains pits, trenches, and vaults for underground waste disposal.
- The TSA, located in the southern section of the facility, is a 58-acre (23-ha) retrievable waste examination and storage area that contains waste stored above ground.

The TSA is dedicated to the temporary storage of contact-handled and remote-handled solid TRU waste. The site includes the Stored Waste Examination Pilot Plant, the air support weather shield, the Drum Venting Facility (where filters are installed in the lids of waste drums to prevent hydrogen buildup), a maintenance shop, the TRU package transporter loading station, Type-I and Type-II storage modules, and the TSA and Retrieval Enclosure. The Advanced Mixed Waste Treatment Facility also is located at the TSA.

The designation recognized under the FFA/CO and CERCLA for the RWMC is WAG 7 (see Figure 1-1).

2.3.7.7 Waste Composition by Facility. Waste produced by the waste generators listed below includes scrap metals, soil, gravel, brick, concrete rubble, sludge, wood, granular carbon, combustible materials slightly contaminated with radionuclides, and a variety of waste associated with research and development and the cleanup of facilities. Descriptions are provided below of the specific composition of the waste produced at the six facilities identified as other waste generators.

- **Auxiliary Reactor Area**—Waste from ARA consists primarily of cleanup waste resulting from past operations and primarily is composed of soil and rubble from gravel, brick, and concrete.
- **Central Facilities Area**—Waste from CFA comes from a variety of the service operations including the laundry, machine shops, maintenance shops, sewage treatment facilities, laboratories, transportation, capital construction, environmental and radiological monitoring, security, medical services, and the cafeteria. It includes scrap metals, rubble, combustibles, and biological waste.

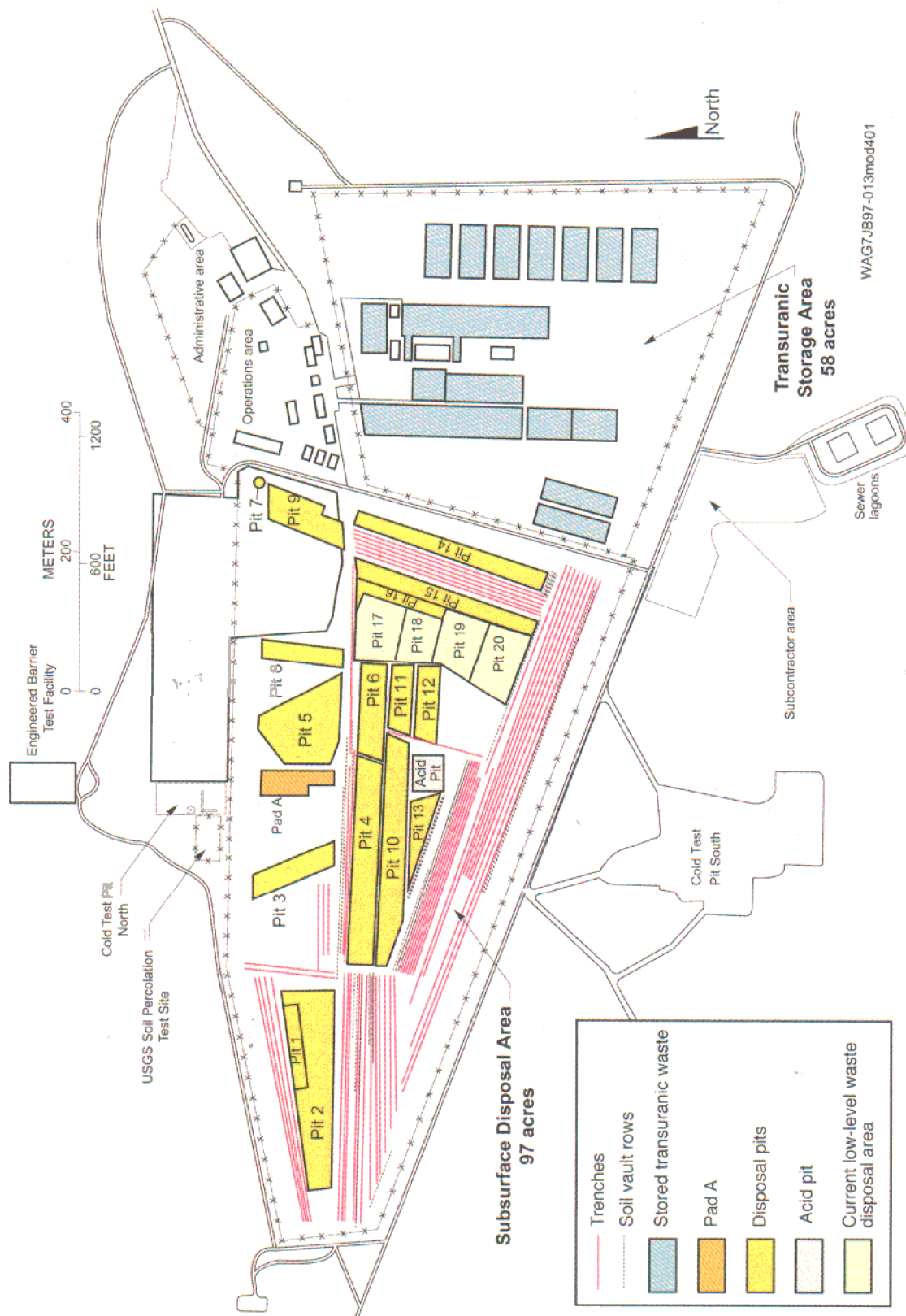


Figure 2-6. Map of the Radioactive Waste Management Complex.

- **Deactivation, Decontamination, and Decommissioning**—Waste generated by D&D&D operations primarily consists of building debris in the form of concrete rubble, brick, wiring, and wood.
- **Power Burst Facility**—Waste from the PBF area was produced from earlier operations of the four SPERT reactors. This waste is primarily metals and combustibles. Some of this waste also comes from operations at the evaporation pond.
- **Waste Reductions Operations Complex**—Most of the waste shipped from WERF to the SDA was generated at other facilities. Treatment of this waste at WERF produced either ash from incineration, metals from the size reduction of large pieces of metal equipment, or compacted metal and wood from the 200-ton compactor.
- **Radioactive Waste Management Complex**—Decontamination of equipment is the primary waste generating process at the RWMC. Cleanup of spills and various treatability studies also contribute on an occasional basis.

Radioactivity for the waste produced by the other waste generators was considered to be insignificant. Therefore, scaling factors were not applied because of the low levels and small volume of waste involved.

2.3.7.8 General Availability of Information. The IWTS database is the main source of data pertaining to the waste generated from other facilities for the time period 1994 to 1999.

2.3.7.9 Data-Collection Approach. The data collection approach required entering the IWTS data information for the other facilities into a spreadsheet and sorting the data by shipment year, destination, generating facility, and waste type to arrive at the total volume of waste and total radioactivity from each facility, waste-container type, and waste type. Waste stream characteristics also were obtained from the IWTS.

2.3.7.10 Descriptions of Waste Streams. The waste produced by the other waste generators from 1994 to 1999 has been categorized into the 28 waste streams listed in Table 2-23.

2.3.7.11 Scaling-Factor Analysis. Curie inventories were so small that attempts to update them with scaling factors were not considered. Upper bounds were calculated using the generic relative standard deviations found in the RPDT (LMITCO 1995a, Section 5.4.3). These generic numbers are derived from EPRI data (EPRI 1987).

2.4 Method for Estimating Contaminant Quantities in Future Waste

Actual waste quantities were used in this supplement for the years 1994 through 1999. No change was made to the estimates for future waste quantities listed in the RPDT.

Table 2-23. Waste streams of other waste generators^a from 1994 to 1999.

Waste Stream Number	Description of Waste
ARA-627-1	Soil, gravel, brick, and concrete rubble
CFA-RWM-1	CFA Sewage Treatment Plant unpainted concrete rubble, drying beds soils, clarifier piping, and trickle filter bricks
CFA-EBR-2	ER Waste Management low-level waste LLW concrete
CFA-EBR-3	ER waste management organization LLW steel
CFA-FPH-1	Any combination of paving, roofing, soil, gravel, brick, and concrete rubble
CFA-691-1	Sludge
CFA-690-1	Metal—stainless steel
D&D-ARA-1	LLW from the decontamination and demolition of the ARA facilities. Waste stream consists primarily of contaminated metal and debris
D&D-CFA-1	Building debris in the form of concrete rubble with some wiring and wood from the demolition of CFA old laundry
D&D-EBRI-1	Waste generated from Experimental Breeder Reactor I demolition activities primarily in the form of metals
D&D-D&D-1	Wood, with some metal generated from the demolition of an unknown building.
PBF-OU5-1	PBF-10 Reactor Area evaporation pond (PBF-733)
PBF-613-1	LLW wood and metal from the Waste Reduction Operations Complex to be direct disposed (i.e., without pretreatment)
PBF-SFD-1	Nonprocessible LLW from the severe fuel damage test
WAG-WG1-01	TSF-17 neutralization pit waste (Operable Unit 1-04)
WAG-WG1-02	WRRTF-04 radioactive liquid tank sludge (solidified)
WG3-313-1	Noncompactible LLW soil samples and Comprehensive Environmental Response, Compensation, and Liability Act material
WG3-308-1	Soils and concrete debris from the Idaho Nuclear Technology and Engineering Center Environmental Control Area-15
WAG-WG7-01	Granular carbon
WAG-WG7-02	Acid pit in situ stabilization treatability study
WMF-610-01	Spill cleanup materials
WMF-601-04	Noncompactible LLW
WMF-602-03	Noncompactible LLW
WMF-ERD-02	Waste returned from laboratory
WER-CMP-1	Compacted waste (combination of glass, plastic, absorbents, cloth, paper, and wood)
WER-INC-1	Incinerated waste (unsolidified ash)
WER-SIZ-1	Sized waste (nonincinerable, noncompactible, and any combination of ferrous and nonferrous metal articles, cloth, paper, plastic, wood, carbon and stainless steel)
WER-WER-1	Repackaged, consolidated waste (low-level wood and metal direct disposed [i.e., without pretreatment], plywood and depleted uranium containers)

ARA = Auxiliary Reactor Area

CFA = Central Facilities Area

ER = environmental restoration

LLW = low-level waste

PBF = Power Burst Facility

a. These waste generators are minor contributors.